December 4, 2003

LICENSEE: Union Electric Company

FACILITY: Callaway Plant

SUBJECT: SUMMARY OF MEETING WITH UNION ELECTRIC COMPANY ON NOVEMBER 12, 2003, TO DISCUSS TWO LICENSEE APPLICATIONS AND POTENTIAL INSTRUMENTATION DIGITAL UPGRADES AT THE PLANT (TAC NOS. MB9875, MB9876, AND MB9879)

A meeting was held on Wednesday, November 12, 2003, between the NRC staff and Union Electric Company, the licensee for the Callaway Plant. The meeting was mutually agreed to by the NRC and the licensee to allow (1) discussions on the licensee's responses to requests for additional information (RAIs) on two applications dated June 27, 2003 (ULNRC-04592 and ULNRC-04868), which proposed changes to the Technical Specifications for the Callaway Plant, Unit 1, and (2) the licensee to present its plans to upgrade instrumentation at the plant. The notice for the meeting was issued on October 28, 2003.

The two license amendment request applications are for proposed modifications to components of the main feedwater and auxiliary feedwater (MFW/AFW) systems, re-analysis of the steam generator tube rupture (SGTR) with overfill accident, and constructing an opening in the secondary shield wall to facilitate maintenance of the replacement steam generators to be installed in Refueling Outage 14 (Fall of 2005).

Enclosure 1 is the list of attendees. Enclosure 2, the handout from the NRC staff, is the agenda for the meeting. Enclosure 3 are the slides handed out by the licensee. Enclosures 4 and 5 are the questions that had previously been e-mailed to the licensee based on the staff's review of the two applications, and the licensee's responses to the questions, respectively. Enclosure 6 is a list of acronyms used in this meeting summary. Enclosure 7 is a list of additional questions raised by the NRC staff because of the discussions on the licensee's responses in Enclosure 5.

The responses to the staff's RAIs were submitted by e-mail from the licensee. In the e-mails, there is a footer that states the following:

> The information contained in this message may be privileged and/or confidential and protected from disclosure. If the reader of this message is not the intended recipient, or an employee or agent responsible for delivering this message to the intended recipient, you are hereby notified that any dissemination, distribution or copying of this communication is strictly prohibited.

The licensee was requested in the meeting, and agreed, to allow the NRC to issue the e-mail responses to the RAIs in the summary for this meeting.

The agenda (Enclosure 2) for the meeting was the following:

- Introduction
- MFW/AFW Mods and SGTR Accident Re-analysis Application (ADAMS ML031950570) - Information Needed by the Staff
  - Richard Eckenrode*
  - Chu-yu Liang*
  - Richard Lobel*
  - Joseph Golla*
- Shield Wall Penetration and LBB Methodology Application (ADAMS ML031950570) - Information Needed by the Staff
  - Roger Pedersen*
  - Mark Hartzman*
  - Pat Patnaik*
- Potential Instrumentation Digital Upgrades for the Callaway Plant
- Adjourn Meeting

(* denotes the NRC reviewer)

In the first two parts of the meeting, the NRC reviewers discussed the responses to their RAIs (Enclosure 5) with the licensee. Clarification of the staff's RAI or the licensee's response to the RAI was discussed in the meeting. The NRC identified any additional information that should be included in the responses to the RAIs that will be submitted by letter to the NRC. The NRC evaluation will be based on the formal responses that are submitted by the licensee.

After lunch, the third part of the meeting on the potential digital upgrades for the Callaway Plant was started. The discussion included a representative from the Wolf Creek Nuclear Operating Corporation (WCNOC) because WCNOC is considering similar upgrades for the instrumentation at the Wolf Creek Generating Station. The licensee's presentation is given in the slides in Enclosure 3. Framatome ANP's slide presentation on diversity and defense-in-depth for the instrumentation digital upgrades was proprietary and, therefore, is not available to the public.

By letter dated November 6, 2003, Framatome ANP submitted an affidavit requesting that information in the presentation material entitled, "Callaway Plant – LSELS, BoP ESFAS, MSFIS Safety DCS Replacement Project – Conceptual Design Overview," to be used at the meeting between the NRC staff and Union Electric Company (licensee for Callaway) on November 12, 2003, concerning potential instrumentation digital upgrades, be withheld from public disclosure pursuant to Title 10 of the Code of *Federal Regulations* (10 CFR) Section 2.790. The letter from the NRC approving this request was issued November 20, 2003. Although the slides used by Framatome ANP in their presentation were not the same as those submitted in the letter dated November 6, 2003, the slides were similar and contained the same material. These slides will not be used as part of any licensing action proposed by the licensee.

The licensee and Framatome completed their presentation and the meeting was adjourned.

/RA/
Jack Donohew, Senior Project Manager, Section 2
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-483

Enclosures:
1. List of Meeting Attendees
2. NRC Staff's Handout for Meeting
3. Licensee's Handout (ADAMS Accession No. ML033220240)
4. Staff's Requests for Additional Information
5. Licensee's Responses to Requests for Additional Information
6. List of Acronyms
7. Additional Questions Raised by the NRC Staff

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Package No.: ML033421470
Meeting Notice No.: ML032960521
ADAMS Accession No.: ML033421469 **NRC-001**

OFFICE	PDIV-2/PM	PDIV-2/LA	PDIV-2/SC
NAME	JDonohew	EPeyton	SDembek
DATE	12/3/2003	12/2/03	12/3/03

DOCUMENT NAME: C:\ORPCheckout\FileNET\ML033421469.wpd
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LIST OF MEETING ATTENDEES

MEETING OF NOVEMBER 12, 2003, WITH CALLAWAY PLANT

NAME	AFFILIATION
J. Donohew	NRC/NRR/PDIV-2
S. Dembek	NRC/NRR/PDIV-2
R. Eckenrode	NRC/NRR/IROB
R. Lobel	NRC/NRR/SPSB
C. Liang	NRC/NRR/SRXB
J. Golla	NRC/NRR/SPLB
M. Hartzman	NRC/NRR/EMEB
C. Doutt	NRC/NRR/SPSB
E. Marinos	NRC/NRR/EEIB
M. Chiramal	NRC/NRR/EEIB
P. Rebstock	NRC/NRR/EEIB
P. Loeser	NRC/NRR/EEIB
M. Waterman	NRC/NRR/EEIB
D. Shafer	Ameren UE
T. Herrmann	Ameren UE
K. Mills	Ameren UE
J. Little	Ameren UE
M. Henry	Ameren UE
P. Bisges	Ameren UE
D. Wingbermuehle	Ameren UE
S. Wideman	WCNOC
W. Eates	WCNOC
L. Colums	Westinghouse
M. Waters	Westinghouse*
M. Watson	Westinghouse*
D. Bhowmick	Westinghouse*
P. Mangano	Framatome
J. Mauck	Framatome
J. Pflugbeul	Framatome
P. Liddle	Framatome

* person participated by phone

Where:

NRC	=	Nuclear Regulatory Commission
NRR	=	Office of Nuclear Reactor Regulation
PDIV-2	=	Project Directorate IV-Section 2
EEIB	=	Materials and Chemical Engineering Branch
EMEB	=	Mechanical and Civil Engineering Branch
SPLB	=	Plant Systems Branch
SPSB	=	Probabilistic Safety Assessment Branch
SRXB	=	Reactor Systems Branch
AmerenUE	=	Ameren Union Electric Company
WCNOC	=	Wolf Creek Nuclear Operating Corporation
Westinghouse	=	Westinghouse Electric Company
Framatome	=	Framatome ANP

ENCLOSURE 1

NRC STAFF'S HANDOUT FOR MEETING

ENCLOSURE 2

AGENDA
NRC/UNION ELECTRIC COMPANY MEETING
CALLAWAY PLANT, UNIT 1

NOVEMBER 12, 2003

- Introduction
- MFW/AFW Mods and SGTR Accident Re-Analysis Application (ADAMS ML031950570)
  Information Needed by Staff
- Shield Wall Penetration and LBB Methodology Application (ADAMS ML031920631)
  Information Needed by Staff
- **PUBLIC QUESTIONS**
- Potential Instrumentation Digital Upgrades for Callaway Plant
- **PUBLIC QUESTIONS**
- Adjourn Meeting

Acronyms:	AFW	auxiliary feedwater
	LBB	leak before break
	MFW	main feedwater
	Mods	modifications
	SGTR	steam generator tube rupture

CALLAWAY MEETINGS ON NOVEMBER 12, 2003

1.	MFW/AFW MODs AND SGTR RE-ANALYSIS LARs		(08:30 - 10:30)
	Richard Eckenrode	08:30 -	(response received)
	Chu-yu Liang	09:00 -	(response received)
	Richard Lobel	09:30 -	(response received)
	Joseph Golla	10:00 - 10:30	(response received)
2.	SHIELD WALL PENETRATION AND LBB METHODOLOGY		(10:30 - noon)
	Roger Pedersen	10:30 -	(response received)
	Mark Hartzman	11:00 -	(no response received)*
	Pat Patnaik	11:30 -	(no response received)*
3.	INSTRUMENTATION DIGITAL UPGRADES		(13:00 - 15:30)
	Angelos Marinos		
	Matt Chiramal		

* The licensee's responses were in fact been received before the meeting

LICENSEE'S HANDOUT FOR NOVEMBER 12, 2003, MEETING

(ADAMS ACCESSION NO. ML033220240)

Three sets of slides:

1. CMP 00-1009A Replacement of Feedwater Isolation Valve (FWIV) Actuators
2. Secondary Shield Wall Door
3. I&C Digital Upgrade Plan/Diversity & Defense in Depth Analysis

ENCLOSURE 3

STAFF'S REQUESTS FOR ADDITIONAL INFORMATION

Technical Review Areas:

A. MFW/AFW Modifications (TAC No. MB9875)
B. Regarding SGTR Accident Re-analysis (TAC No. MB9876)
C. Regarding Opening in Secondary Shield Wall (TAC No. MB9879)
D. Regarding LBB Methodology for Opening in Secondary Shield Wall (TAC No. MB9879)

ENCLOSURE 4

A. MFW/AFW Modifications (TAC No. MB9875)

The license amendment application from the licensee was submitted in the letter dated June 27, 2003 (ULNRC-04592).

A.1 Plant Systems Review:

1. Was a failure modes and effects analysis, or similar analysis, performed for the MFIV actuator modification and the motor driven auxiliary feedwater (AFW) pump discharge check valve replacement with "ARC" valves? If so, discuss the results.

2. Discuss if a MFIV will be able to close in the event of a feed water line break (FWLB). Address if the new MFIV actuators operate on system pressure and the possibility for a large FWLB to cause a loss of system pressure rapidly enough so that the associated MFIV does not have enough system pressure to close. The closed MFIV is needed to act as a pressure boundary for AFW injection.

A.2 Containment Safety Assessment Review:

1. Given the increased MFIV closure time and the resulting longer steam generator dryout time for the MFW/AFW systems modifications, discuss the differences in analysis assumptions that result in the Bechtel containment analysis of containment pressure and temperature to remain bounding with respect to the CONTEMPT containment analysis.

2. Describe the assumptions used in the CONTEMPT containment calculations. Discuss if the assumptions included the temperature flash assumption and if the Tagami and Uchida heat transfer correlations were used. Address what conservatisms are included in the calculation of structural heat sink areas and coatings, and in the value of the containment volume.

3. Explain how the generic calculations of mass and energy release can be independent of feedwater (FW) assumptions and discuss if this is related to the discussion of Section 3.1.5 of WCAP-8822.

4. It is stated that the peak containment pressure calculated by CONTEMPT remains below the Bechtel analysis for a steam line break with an increase in stroke time of the MFIVs and for an increase in the maximum auxiliary feedwater (AFW) flow due to the ARC valves when the cases are considered separately. This is stated in the last paragraph of the Main Steam Line Break Mass and Energy Release Analysis section (starting page 21 of 33) of Attachment 2. Discuss an verify that when both effects are considered together the peak containment pressure remains below the Bechtel analysis for a steam line break.

B. Regarding SGTR Accident Re-analysis (TAC No. MB9876)

The license amendment application from the licensee was submitted in the letter dated June 27, 2003 (ULNRC-04592).

B.1 Human Factors Considerations:

In reviewing the application dated June 27, 2003, the staff has determined it needs the following information concerning the times listed in the table on page 17 of 33 of Attachment 2, "Evaluation," to the letter:

1. The paragraph just below the table states that "Simulated control room exercises were performed in 2003 for this accident. The exercises have demonstrated that the operator action times that serve as inputs to the thermal-hydraulic analysis have increased above the times originally analyzed in the SGTR with overfill analysis presented to the NRC in [letter] ULNRC-1518, dated May 27, 1987 ..." The sentences in the application may mean that the times in the previous analysis were increased to those listed in the table solely to meet the increased operator action times observed in simulated control room exercises in 2003. Discuss this and explain what the times in the table are based upon.

2. The NRC staff's evaluation dated March 30, 1987, of the Westinghouse Owners group WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," stipulated plant-specific criteria for assessing operator action times in the event of an SGTR. Address the criteria as updated below:

a. Provide simulator and emergency operating procedure training related to a potential SGTR.

b. Using typical control room staff as participants in demonstration runs, show that the operator action times assumed in the SGTR analysis are realistic and achievable.

c. Complete demonstration runs to show that the postulated SGTR accident can be mitigated within a period of time compatible with overfill prevention, using design basis assumptions regarding available equipment and its impact on operator response times. All control room crews should demonstrate a response time which is less than the operator response time assumed in the analysis for the accident.

d. Describe the means the emergency operating procedures specify for identifying the steam generator (SG) with the ruptured tube, provide the expected time period for determining that SG, and discuss the effects on the duration of the accident.

3. Below is an additional question based on the following statements given in the licensee's responses to the questions above and shown in **bold** in Section B.1 of Enclosure 5:

"It should be noted that the Callaway SGTR analysis is not performed using the methodology described in WCAP-10698. The Callaway SGTR analysis is based on the SNUPPS methodology which was originally developed for Callaway and Wolf Creek."

> "Overfill prevention is not demonstrated at Callaway. The analysis and operator action times are commensurate with mitigation of the consequences of an overfill event."

Discuss why the SGTR analysis and operator actions are set for mitigation of the consequences of an overfill event instead of being set for prevention of the overfill event (i.e., WCAP-10698).

B.2 Accident Radiological Consequences

As explained in Regulatory Information Summary 2001-19, “Deficiencies in the Documentation of Design Basis Radiological Analyses Submitted in Conjunction with License Amendment Requests,” the NRC staff bases its finding on the acceptability of an amendment on its assessment of the licensee’s analysis. For the NRC staff to make an acceptable finding, the licensee must provide adequate information regarding analysis assumptions, inputs, and methods in the submittal. If any of the requested information was previously docketed for Callaway, the licensee is requested to provide the specific citation in its response.

1. The licensee did not address the impact of the proposed changes on the ability of the control room habitability systems to maintain doses to operators within the criteria of 10 CFR Part 50, Appendix A, General Design Criteria 19. The staff notes that the analysis of record overfill case thyroid dose (pre-incident spike) at the EAB was 24 rem and that this had increased to 46 rem in the new overfill case. Given the significant increase in the EAB dose, the staff suspects that the control room dose would have similarly increased. Respond to Question a or b below as applicable.

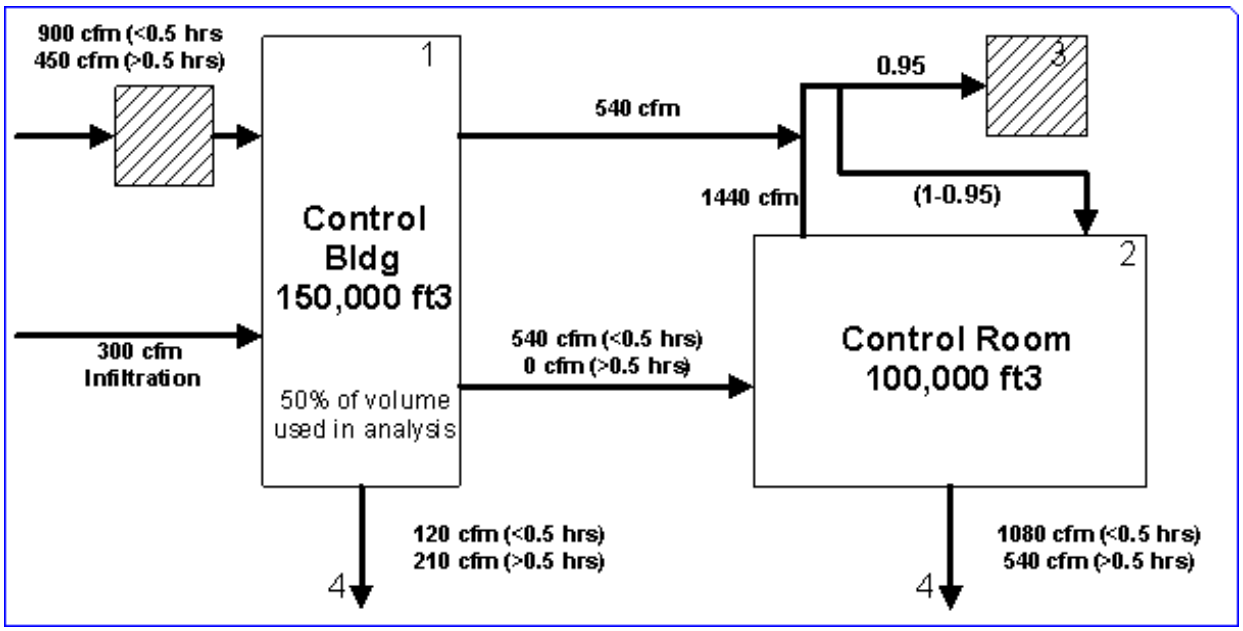
    a. Provide a description of the assumptions, inputs, methods and results of the evaluation that demonstrates that GDC-19 will continue to be met.

    b. If the licensee has not evaluated the control room dose but is relying on the dose being bounded by that determined for another accident, provide a justification that addresses the considerations in Paragraphs 7a through 7d of RIS 2001-19, as applicable.

    In either case, please provide the information requested in item 1.a of Generic Letter 2003-01, as it applies to the SGTR with overfill, in your response. If you have already docketed your response to GL 2003-01, provide a citation to that response.

2. Figures 15.6-3P, 15.6-3.2d, and 15.6-3.2h of the submittal provide data for the intact steam generators (SGs). Discuss if the data represents each intact SG or the total for all intact SGs.

3. The figure below represents the staff’s interpretation of appropriate modeling of your control building and control room. Node 3 is the recirculation filter; node 4 is a “sink.” Discuss the staff's interpretation and confirm if the staff’s understanding is correct. In particular, discuss if the expected re-alignment will occur at 30 minutes for this event and provide the basis for this conclusion.



4. Discuss (1) the iodine appearance rates for I-131 to I-135, in Ci/hr, to which the multiplier of 335 will be applied and (2) the assumed duration of the accident induced spike.

In reviewing the licensee's application, the staff had the following comments on statements made by the licensee. First, with regard to the conclusion that 10 CFR 50.67 applied to this amendment, the staff believes that the requirements of 10 CFR 50.67 do not apply to this amendment request. This is based on the definition of "source term" in 10 CFR 50.2 and the statements of consideration for the final 10 CFR 50.67 rule (63 FR 71990 dated December 23, 1999). Second, the licensee has requested staff approval to use Regulatory Guide (RG) 1.195 for other licensing basis dose applications. The staff believes that 10 CFR 50.59 already provides the licensee with an adequate mechanism to implement the guidance of RG 1.195. The guide provides methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations; however, the guide also contains alternative methods that must be considered on a case-by-case. The staff considers blanket approval of the use of RG 1.195 would confer approval for each of these case-by-case situations, most of which are not considered relevant to the technical specification changes requested or to the re-analysis of the SGTR with overfill. If the licensee believes that the staff has misunderstood its rational in these two areas, the licensee should provide further explanation of its position.

B.3 Reactor Systems Review

1. It is stated in the submittal that the SGTR re-analysis is performed with revised operator action times, inputs, and assumptions that are consistent with the current plant configuration and operation. Tabulate these changes and provide a justification for each change.

2. Discuss the computer codes used in the thermal-hydraulic analysis of the SGTR event. If any of the computer codes are not approved by the NRC, provide a justification for their use in the analysis.

3. Provide any revised emergency operating procedure (EOP) steps (in E-2 and E-3) that are related to the actions required for isolation of auxiliary feedwater (AFW) flow to the failed steam generator (SG).

4. For the case of an AFW control valve failing to its open position, discuss what operator actions are needed to isolate the failed SG from continued AFW flow injection (i.e., what backup valve needs to close and its location), including where these actions are performed.

5. Discuss the integrity of the main steam line for SG overfill (i.e., with the entire main steam line up to the MSIV filled with water).

6. For the main steam line break (MSLB), discuss the effect of the proposed longer closure time (i.e., 15 seconds) of the main feedwater isolation valves (MFIVs).

7. Following the SG overfill, it is assumed that the safety valve associated with the failed SG is stuck open with an effective flow area equal to 5% of the total safety valve flow area. Discuss the basis for the assumed effective flow area value.

8. In the sequence of events for a SGTR with overfill, it is indicated that the operator actions to terminate AFW flow from the turbine driven AFW (TDAFW) pump to the failed SG will be completed within 10 minutes following the event initiation. Discuss why these actions are expected to be completed within 10 minutes while the operator actions to isolate AFW flow from MDAFW pumps are expected to be completed within 20 minutes.

9. Figure 15.6-3P, "Feedwater Flow Rate," of the proposed FSAR page changes, does not appear to be consistent with the time assumed for closure of the MFIVs (i.e., 15 seconds). Discuss how the FSAR figure is consistent with the assumed MFIV closure time.

10. Discuss why the change to the MFIV closure time from 5 seconds to 15 seconds does not affect other events for which re-analyses of these event should be performed to support the proposed Technical Specification changes.

C. Regarding Opening in Secondary Shield Wall (TAC No. MB9879)

The following is an RAI for the license amendment application dated June 27, 2003 (ULNRC-04868), related to an opening in the secondary shield wall: On review of the Callaway drawings (in the FSAR and the additional drawing provided by the licensee to the NRC project manager), it has been identified that the proposed opening in the shield wall, which is depicted in drawing no. C-2S2977, may not be consistent with the following statement by the licensee on Page 5, second paragraph, of Attachment 1 to the application: "In order to preclude radiation streaming and dose resulting from creating the opening in the secondary shield cubical wall, alternative shielding will be applied to the opening and access control entryway to limit radiation doses consistent with maintaining them as low as [is] reasonably achievable (ALARA)." The alternative shielding indicated on the drawings does not appear to "preclude streaming" for the following reasons:

1. The shield does not cover the entire height of the opening. Approximately four square feet of opening is unshielded.

2. No shielding is provided on the top, or the deck, of the security cage to prevent radiation (penetrating the secondary shield wall opening at a up or down slant angle) from streaming into accessible spaces above and below the security cage.

3. The "alternative" lead shield that is included in the design will provide only approximately three orders of magnitude less shielding (one tenth thickness of lead verses about four tenth thicknesses of concrete).

Therefore,

1. Explain how the alternative shielding is as effective as the original shield design, or

2.a) Calculate the expected increase in radiation dose rates outside the secondary shield cubical resulting from this modification.

2.b) Estimate the increase in dose expected for workers accessing these areas during periods of reactor shutdown and power operations and show that this is ALARA.

2.c) Verify that the modification does not create a Very High Radiation Area, as defined in 10 Part 20, or describe the "additional measures" required by 10 CFR 20.1602.

D. Regarding LBB Methodology for Opening in Secondary Shield Wall (TAC No. MB9879)

The license amendment application from the licensee was submitted in the letter dated June 27, 2003 (ULNRC-04868).

D.1 Materials Branch Review

1. Discuss the rationale for not considering the potential effect of thermal stratification, cycling and striping (TASCS) in the horizontal sections of RHR and accumulator piping discussed in the subject WCAP topical reports (provided in the application dated June 27, 2003) due to potential leakage of hot fluids past the isolation valve or heat transfer across the isolation valve, including, in particular, the startup of the systems. The discussion should also address the contribution to thermal fatigue due to TASCS in the above cases and will the nodal location of maximum stress change as a result of the above.

D.2 Mechanical Engineering Branch Review

The following are requests for information on the following three Westinghouse topical reports submitted by the licensee for the LBB methodology in the application:

- WCAP-15983-P
- WCAP-16019-P
- WCAP-16020-P

A. The following are questions related to the staff's review of WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," dated February 2003:

1. Provide assurance that the internal loads used in the calculations were determined based on the as-built configuration of the surge line.

2. Provide assurance that the wall thickness of all components of the surge line meets the minimum ASME Code, Section III, Class 1, wall thickness requirements.

3. Discuss compliance with the snubber surveillance requirements of the Callaway Technical Requirements Manual (TRM) to provide assurance that snubber failure rate are acceptably low.

4. Section 4.2 of the WCAP defines "TH" as the "Applicable Thermal Expansion Load (Normal or Stratified)." Provide clarification indicating if the "normal thermal expansion load" includes both axial and bending loads, and if the thermal stratification also includes axial and bending loads. Provide justification why normal and stratified thermal loads are not additive.

5. In Section 4.2 of the WCAP, the applied loads for crack stability analysis include loads due to seismic anchor motion (SAM) and the leak before break (LBB) margin is stated to be reduced to 1.0. SRP 3.6.3 indicates that the margin of 1.4 can be reduced to 1.0 if these loads are combined absolutely with the safe shutdown earthquake (SSE) load and the individual operating loads. Provide verification that the SAM loads were combined absolutely with the SSE and the operating loads, as specified in the SRP 3.6.3.

6. In Table 4-2 of the WCAP, which shows Cases A through G for normal and faulted loading cases for LBB evaluations, provide the justification for not including normal thermal expansion loads in the load combination Case E.

7. For Node 3030 in Table 4-4 of the WCAP, which provides a summary of LBB loads and stresses, provide a table for Cases A through G of Table 4-2 showing the individual force and moment components due to pressure, deadweight, thermal expansion, thermal stratification, SSE inertia and SSE SAM.

8. In Section 4.5 of the WCAP, provide the minimum wall thickness at the weld counterbore used in the analysis.

9. Provide the basis or a reference for Equation 5-3 in Section 5.2.2 of the WCAP.

10. In support of the statement in Section 5.2.3 of the WCAP, that "The leak rates were calculated using the normal operating loads at the governing location identified in Section 4.0," provide a representative detailed leak rate calculation, including the calculation of the crack opening area, for the postulated through-wall circumferential crack at Node 3030.

11. The fatigue crack growth analysis for the pressurizer surge line was performed at the same location where the maximum ASME Code, Section III, Class 1, cumulative usage factor was previously calculated under the effects of thermal stratification. Provide the specific location on the pressurizer surge line, and the value of the maximum cumulative usage factor used in the analysis.

12. In Section 6.2 of the WCAP, the term "stress cuts" is used in the statement that "Fatigue crack growth analyses were carried out along five stress cuts ..." Discuss what a "stress cuts" is and provide an explanation of the manner in which it is used in the fatigue crack growth analyses.

B. For WCAP-16019-P, "Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," the questions are the following:

1. Provide the ASME Code, Section III, Class 1, cumulative usage factors at Nodes 3020 (Accumulator Loop 2), 3120 (Accumulator Loop 2), and 3295 (Accumulator Loop 3).

2. Provide assurance that the internal loads used in the calculations were determined based on the as-built configuration of the accumulator lines.

3. Provide assurance that the wall thicknesses of all components of the accumulator lines meet the minimum ASME Code, Section III, Class 1, wall thickness requirements.

4. Discuss compliance with the snubber surveillance requirements of the Callaway Technical Requirements Manual (TRM) to provide assurance that the snubber failure rate on the accumulator lines is acceptably low.

5. In Section 4.2 of the WCAP, the applied loads for crack stability analysis include loads due to the safe shutdown earthquake (SSE). Provide verification that SSE loads include both inertia

and seismic anchor motion loads, and that these are combined absolutely, as specified in the Standard Review Plan (SRP) 3.6.3.

6. Provide detailed justification for not including operating basis earthquake (OBE) loads in the fatigue crack growth analysis

C. For WCAP-16020-P, "Technical Justification for Eliminating 12" Residual Heat Removal (RHR) Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," the questions are the following:

1. Provide the ASME Code, Section III, Class 1, cumulative usage factors at Nodes 3285 (RHR Line Loop 1) and 3020 (RHR Line Loop 4).

2. Provide assurance that the internal loads used in the calculations were determined based on the as-built configuration of the residual heat removal (RHR) lines.

3. Provide assurance that the wall thicknesses of all components of the RHR lines meet the minimum ASME Code, Section III, Class 1, wall thickness requirements.

4. Discuss compliance with the snubber surveillance requirements of the Callaway TRM to provide assurance that snubber failure rate are acceptably low.

5. In Paragraph 4.2, the applied loads for crack stability analysis include loads due to SSE. Provide verification that SSE loads include both inertia and seismic anchor motion loads, and that these are combined absolutely, as specified in the SRP 3.6.3.

6. Provide detailed justification for not including OBE loads in the fatigue crack growth analysis

LICENSEE'S RESPONSES TO REQUESTS FOR ADDITIONAL INFORMATION

Technical Review Areas:

A. MFW/AFW Modifications (TAC No. MB9875)
B. Regarding SGTR Accident Re-analysis (TAC No. MB9876)
C. Regarding Opening in Secondary Shield Wall (TAC No. MB9879)
D. Regarding LBB Methodology for Opening in Secondary Shield Wall (TAC No. MB9879)

ENCLOSURE 5

A. MFW/AFW Modifications (TAC No. MB9875)

A.1 Plant Systems Review:

NRC Question 1:

Was a failure modes and effects analysis, or similar analysis, performed for the MFIV actuator modification and the motor driven auxiliary feedwater (AFW) pump discharge check valve replacement with "ARC" valves? If so, discuss the results.

AmerenUE Response:

PART A. FAILURE ANALYSIS FOR MFIV ACTUATOR REPLACEMENT

The safety related function of the MFIV actuator is to close an MFIV in less than or equal to 15 seconds. Each actuator has two actuation trains capable of performing this function. The licensing basis for an MFIV actuator is that a single failure of any active component cannot prevent the actuator from performing its safety function. The following failure analysis discusses the actuator operation considering all possible failures. Along with the discussion below, refer to Attachment 8 of the license amendment request submittal (ULNRC-04592) for the MFIV Actuator Diagram for the failure analysis of the system medium actuator.

- Failure of 'A' ('B') Train ESFAS or ‘A’ (‘B’) Train MSFIS to Actuate

If the 'A' ('B') train of ESFAS or MSFIS fails to actuate, the associated UPC solenoid valves MV1 and MV3 (MV2 and MV4) will remain in an energized state. The associated LPC solenoid valve MV5 (MV6) will remain in a de-energized state. The solenoid valves in the opposite train, MV2 and MV4 (MV1 and MV3), will still de-energize, directing feedwater to the UPC. Both LPC solenoids will remain in a de-energized state (vented position) until the MFIV is closed. After a 60 second time delay, the actuated train LPC solenoid valve MV6 (MV5) will go to an energized state (closed or pressurized position). Under these conditions, feedwater will be vented through MV1 (MV2), while the opposite train solenoids, MV2 and MV4 (MV1 and MV3) will route feedwater to the UPC. The exhaust port from MV1 (MV2), however, is equipped with a 2 mm orifice sized to limit exhaust flow to an acceptable level. In addition, feedwater will be vented through either MV5 (MV6), whichever solenoid valve remains de-energized after 60 seconds. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that the MFIV will close in the required 15 seconds under this condition. It was also found, however, that up to 78 lbm/min (~10 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoids.

- Inadvertent Actuation of 'A' ('B') Train ESFAS or ‘A’ (‘B’) Train MSFIS

If the 'A' ('B') train of ESFAS or MSFIS inadvertently actuates, the associated UPC solenoid valves MV1 and MV3 (MV2 and MV4) will go to a de-energized state. The associated LPC solenoid valve MV5 (MV6) will remain in a de-energized state. The solenoid valves in the opposite train, MV2 and MV4 (MV1 and MV3), will remain energized, and the opposite train LPC solenoid valve, MV6 (MV5) will remain de-energized. The de-energized UPC solenoid

valves, MV1 and MV3 (MV2 and MV4), will direct feedwater to the UPC. The de-energized LPC solenoid valves MV5 and MV6 will remain in a de-energized condition (vented position) until the MFIV is closed. After a 60 second time delay, the actuated train LPC solenoid valve MV5 (MV6) will go to an energized state (closed or pressurized position). Under these conditions, feedwater will be vented through MV2 (MV1), while the actuated train solenoids, MV1 and MV3 (MV2 and MV4) will route feedwater to the UPC. The exhaust port from MV2 (MV1), however, is equipped with an orifice sized to limit exhaust flow to an acceptable level. In addition, feedwater will be vented through either MV5 (MV6), whichever solenoid valve remains de-energized after 60 seconds. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that the MFIV will close in the required 15 seconds under this condition. It was also found, however, that up to 78 lbm/min (~10 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoids.

- Solenoid MV1 (MV2) Fails in the Energized State (Vented Position)

MV1 (MV2) is a three-way solenoid valve, which vents the UPC when energized and directs feedwater to the UPC when de-energized. Should the MFIV receive a close signal and MV1 (MV2) fails to de-energize, MV1 (MV2) will remain in the vented position. Under these conditions, feedwater will be vented through MV1 (MV2), while the opposite train solenoids, MV2 and MV4 (MV1 and MV3) will route feedwater to the UPC. Since the exhaust ports are connected, a portion of the feedwater from the opposite train will vent through MV1 (MV2). The exhaust port from MV1 (MV2), however, is equipped with an orifice sized to limit exhaust flow to an acceptable level. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that the MFIV will close in the required 15 seconds under this condition. It was also found, however, that up to 45 lbm/min (~6 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoid.

- Solenoid MV1 (MV2) Fails in the De-energized State (Pressurized Position)

This is the safe position, and will not adversely impact the ability of the actuator to close the MFIV in the required 15 seconds. If the MFIV is open and MV1 (MV2) de-energizes, feedwater will still be isolated by MV3 (MV4). Therefore, this single failure will not prevent the MFIV from closing in the required time or cause the MFIV to close creating a Reactor Trip.

- Solenoid MV3 (MV4) Fails in the Energized State (Closed Position)

MV3 (MV4) is a two-way solenoid valve, which isolates feedwater from the inlet to MV1 (MV2) when energized and directs feedwater to MV1 (MV2) when de-energized. Should the MFIV receive a close signal and MV3 (MV4) fails to de-energize, feedwater will still be directed to the UPC through the opposite train, MV2 and MV4 (MV1 and MV3). Therefore, this single failure will not prevent the MFIV from closing in 15 seconds or cause the MFIV to close creating a Reactor Trip.

- Solenoid MV3 (MV4) Fails in the De-energized State (Open Position)

This is the safe position, and will not adversely impact the ability of the MFIV actuator to close the valve in the required 15 seconds. If the MFIV is open and MV3 (MV4) de-energizes,

feedwater will still be isolated from the UPC by MV1 (MV2). Therefore, this single failure will not prevent the MFIV from closing or cause the MFIV to close creating a Reactor Trip.

- Solenoid MV5 Fails in the Energized State (Closed Position)

MV5 is a two-way solenoid valve, which isolates the LPC vent path when energized and provides a vent path for the LPC when de-energized. Should the MFIV receive a close signal and MV5 fails in the energized state (closed position), the LPC will still be vented through the opposite train, MV6. The MFIV will still close in the required 15 seconds with MV5 in the energized state (closed position).

- Solenoid MV5 Fails in the De-energized State (Open Position)

This is the primary safe position, and will not adversely impact the ability of the MFIV actuator to close the valve in the required 15 seconds. If the MFIV is open and MV5 fails in the de-energized state (open position), the LPC will be vented through both MV5 and MV6. Therefore, this single failure will not prevent the MFIV from closing in the required time frame. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that up to 33 lbm/min (~4 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoid.

- Solenoid MV6 Fails in the Energized State (Pressurized Position)

MV6 is a three-way solenoid valve, which directs feedwater to the LPC when energized and provides a vent path for the LPC when de-energized. Should the MFIV receive a close signal and MV6 fails in the energized state (pressurized position), feedwater will be directed to the LPC if the system is pressurized. The LPC will still be vented through the opposite train solenoid, MV5. The inlet port to MV6, however, is equipped with an orifice sized to limit inlet flow to an acceptable level. The MFIV will still close in the required 15 seconds with MV6 in the energized state (pressurized position).

- Solenoid MV6 Fails in the De-energized State (Vented Position)

This is the primary safe position, and will not adversely impact the ability of the MFIV actuator to close the valve in the required 15 seconds. If the MFIV is open and MV6 fails in the de-energized state (vented position), the LPC will be vented through both MV5 and MV6. Therefore, this single failure will not prevent the MFIV from closing in the required time frame. Through discussion with the valve manufacturer, Control Components Inc. (CCI), and subsequent testing, it has been confirmed that up to 33 lbm/min (~4 gpm) of process fluid per actuator, could be diverted from the system back to the condenser through the failed solenoid.

- Loss of Lower Piston Chamber Vent Path

In order to perform the safety function of closing the MFIV in the required 15 seconds, the lower piston chamber must be vented. This is accomplished by providing two redundant vent paths through two LPC solenoid valves, MV5 and MV6, which are then tied to a common header. Two parallel vent paths are then provided from each MFIV vent header. The normal and preferred vent path is back to the condenser. A redundant vent path to a rupture disk, which discharges to an equipment/floor drain in Area 5, is also provided. If the non-safety-related

path fails and the MFIV receives a close signal, once the LPC becomes pressurized, the rupture disk will break providing a vent path, ensuring the MFIV will close in the required 15 seconds.

- Dual Electrical Train Failure

In this design the separation of trains is on the order of one half inch where the wiring comes together on the switches. However, this design complies with the regulatory requirements by providing an insulating barrier for separation by using switches that have been qualified, and by providing a failure modes and effects analysis.

The separation barrier medium is a high temperature ceramic based insulating material. It will not burn. The published operating temperature is 2200 F for extended periods. Further, the sleeving provides electrical insulating properties of high electrical resistance at elevated temperatures, low shrinkage and low moisture absorption characteristics for an excellent electrical insulator.

The power supplies that feed the remote contacts for MSFIS are an ungrounded 48 VDC supply.

- Failures Related to Fast Close Switches

Short to ground of all conductors – This scenario will not cause a power supply failure, since the power supplies are floating with respect to ground. It is possible that a fast close would be initiated, but that is the safeguard position of the valve and conservative.

Short together of all conductors – Same result as explained above.

Fire inside of switch cubicle – This scenario is not credible, since there are no heat sources. Shorting the 48 volt power supply to ground will not draw any currents, because the power supply is floating. Further, the fast close inputs to MSFIS only draw about 20 ma, which is about 1 watt of power dissipated in the MSFIS cabinets. This power is not enough heat to postulate a fire hazard at the switches.

Fire outside of cubicle – This scenario is the same threat as prior to the modification.

Switch breaks off – If the switch breaks off, the only possible outcome would be to cause a fast close on one or both trains of MSFIS, which is the safeguard position.

- Failures Related to Open/Close Switches

Fire in hand switch resistor deck – It is conceivable to have a fire relating to the indicating lights on these switches. In this case the insulation on the opposite train wiring is protected by the high temperature sleeving. Even if the insulation did melt, the sleeving would provide the required electrical separation.

In the case of hypothesizing a complete failure of components, see the following discussion on shorting to ground.

Short to ground – Since the 48 volt power supplies of both trains are floating, there would be no safety consequence of shorting all wires on the switch together and to ground. An open command or a normal close command could result. However, these switches do not have a safety function; the safety function is provided by automatic actuation signals and the manual Fast Close switches. The safety functions override the logic and inputs from the switch.

Switch breaks off – The only possible scenario would be to create an open or close signal to one or both trains of a single valve. This switch has no safety function. The safety functions always override any inputs from this switch.

- Common Mode Software Failure (CMSF)

Actuation control for the MFIVs is accomplished by the Mainsteam and Feedwater Isolation System (MSFIS), which is a Programmable Logic Controller (PLC)-based digital control system. Although the existing software will remain largely unmodified, one new module will be added to perform the new MFIV actuation logic. A common mode software failure (CMSF) could exist if both trains of PLCs have a simultaneous software malfunction and /or fault. As stated in the Safety Evaluation for Callaway License Amendment 117 (dated October 1, 1996), the possibility of a CMSF is reduced to a very low probability due to the high quality established throughout the software design process. Based on the simplicity of the new actuator design and the extent of the V&V performed on the new software module by the Developer, common mode software failures are no more likely with the new design than they were with the existing design.

Failure Occurs During Normal Operation

Logic Output Failure State			***Solenoid Output Failure State***				
A	**B**	**C**	**A**	**B**	**C**	**Single Train Outcome**	**Common Mode Outcome**
0	0	0	1	1	1	Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
0	0	1	1	1	0	Sol A&B continue to isolate fluid to the upper piston chamber (UPC). The lower piston chamber (LPC) is vented through Sol C.. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
0	1	0	1	0	1	Sol A continues to isolate fluid to the UPC. Sol C energizes to block the lower piston vent path. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
0	1	1	1	0	0	Sol A continues to isolate fluid to the UPC. Sol C deenergizes to vent the LPC. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
1	0	0	0	1	1	Sol B continues to isolate fluid to the UPC. Sol C energizes to block the lower piston vent path. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above
1	0	1	0	1	0	Sol B continues to isolate fluid to the UPC. Sol C deenergizes to vent the LPC. Train unable to accomplish safety function. Other train closes valve within 15 seconds.	See discussion above

Notes:

1) For the purposes of this evaluation, it is assumed that the PLC's fail in the output state shown. A logic output of ‘1’ energizes the actuation relay and ‘0’ deenergizes the actuation relay. A & B solenoids are energized when their actuation relays are deenergized, and C solenoid is energized with an energized actuation relay.

2) Solenoid A corresponds to MV1 or MV2, Solenoid B corresponds to MV3 or MV4 and Solenoid C corresponds to MV5 or MV6.

Failure Occurs During Normal Operation							
Logic Output Failure State			***Solenoid Output Failure State***				
A	**B**	**C**	**A**	**B**	**C**	**Single Train Outcome**	**Common Mode Outcome**
1	1	0	0	0	1	Sol A&B deenergizes to direct process fluid to the (UPC). Sol C energizes to block the lower piston vent path. MFIV closes valve within 15 seconds.	See discussion above
1	1	1	0	0	0	The deenergized Sol A&B direct process fluid to the UPC. Sol C vents the LPC. MFIV closes within 15 seconds.	See discussion above

Notes:

1) For the purposes of this evaluation, it is assumed that the PLC's fail in the output state shown. A logic output of '1' energizes the actuation relay and '0' deenergizes the actuation relay. A & B solenoids are energized when their actuation relays are deenergized, and C solenoid is energized with an energized actuation relay.

2) Solenoid A corresponds to MV1 or MV2, Solenoid B corresponds to MV3 or MV4 and Solenoid C corresponds to MV5 or MV6.

<u>AmerenUE Response</u>: (Continued, A.1 Plant Systems Review)

PART B. FAILURE ANALYSIS FOR AFW PUMP DISCHARGE "ARC" VALVE REPLACEMENT

The failure modes and effects analysis (FMEA) was considered for the modification to replace the MDAFP Discharge Check Valve with the automatic recirculation control check (ARC) valve. In summary, the only change between the existing system with a swing style check valve plus the recirculation line orifice and the replacement ARC valve is the interaction between the two performed by the armature in the replacement ARC valve. In the unlikely event that the armature that connects recirculation flow control and the lifting check disc were to fail (break), the result would be that the recirculation line of the ARC valve would fail open. When the valve fails to the open position, the system returns to the current configuration installed at Callaway, with flow through the recirculation line being continuous, but restricted by the bypass pressure reducer orifice of the ARC valve.

<u>NRC Question 2</u>:

Discuss if a MFIV will be able to close in the event of a feed water line break (FWLB). Address if the new MFIV actuators operate on system pressure and the possibility for a large FWLB to cause a loss of system pressure rapidly enough so that the associated MFIV does not have enough system pressure to close. The closed MFIV is needed to act as a pressure boundary for AFW injection.

<u>AmerenUE Response</u>:

Feedwater isolation valve closure delays are not explicitly modeled in the loss of normal feedwater (LONF) or loss of AC power (LOAC) analyses. The assumed time for AFW delivery to the steam generators, which accounts for system actuation and piping purge delays, implies feedwater isolation since FWIV closure provides the pressure boundary for AFW injection into the steam generators. With regard to the feedwater line break (FWLB) analysis, a FWIV closure time of 68.2 seconds is listed in FSAR Table 15.2-1 and that time is being increased by 10 seconds in this amendment. However, the FWLB analysis is performed in a fashion similar to the LONF/LOAC analyses. As long as FWIV closure occurs within the assumed 60-second AFW actuation delay time, the results of these primary side heatup analyses are not impacted.

A FWLB could potentially result in a rapid secondary side depressurization down to a containment pressure greater than 0 psig. Following assembly of each MFIV system medium actuator, a hot functional test was performed using Callaway's spare MFIV body. Through this testing it was found the MFIV would close within 60 seconds with as little as 0 psig of secondary side system pressure. Therefore, the FWIVs will always close within 60 seconds in response to any LONF/LOAC or FWLB event.

The increased MFIV stroke time (15 seconds) has been evaluated for the MSLB core response and containment P/T analyses with no resultant impact on the conclusions of those analyses. For the MSLB core response analyses, main feedwater isolation is credited to limit the cooldown of the RCS. For the MSLB containment analyses, main feedwater isolation is

credited to limit the main feedwater mass and energy release to containment. In both cases, it is conservative to assume main feedwater flow is maintained until feedwater isolation occurs. All conditions where main feedwater flow is maintained to the steam generators result in secondary side system pressures well in excess of 90 psig. Following assembly of each MFIV system medium actuator, a hot functional test was performed using Callaway's spare MFIV body. Through this testing it was found the MFIV would close within 15 seconds with as little as 90 psig of secondary side system pressure, using cold water (< 250°F) in the system. This testing also found the MFIV actuators would stroke much quicker when hot water (> 250°F) was used. Therefore, the FWIVs will always close within 15 seconds in response to any MSLB event.

A.2 Containment Safety Assessment Review:

NRC Question 1:

Given the increased MFIV closure time and the resulting longer steam generator dryout time for the MFW/AFW systems modifications, discuss the differences in analysis assumptions that result in the Bechtel containment analysis of containment pressure and temperature to remain bounding with respect to the CONTEMPT containment analysis.

AmerenUE Response:

Two input assumptions have been revised since the original Bechtel containment analysis was performed. These are

Condensate re-vaporization

Time assumed for operator action to isolate Aux Feedwater to the affected steam generator

The original Bechtel analysis did not credit re-vaporization. Re-analysis first performed and incorporated into Callaway's FSAR as a part of Callaway's power uprating incorporated a credit of 8% re-vaporization, as allowed by NUREG 0588. The 8% revaporization is discussed in Section 6.2.1.4 of Callaway's FSAR.

The original Bechtel analysis assumed that operators did not isolate Aux Feedwater to the affected steam generator until 1800 seconds. However, the original FSAR Sections 10.4.9 (Auxiliary Feedwater System) and 15.0.13 (Operator Actions) stated that during a Main Steam Line Break, auxiliary feedwater to the faulted steam generator can be terminated within 10 minutes. FSAR Section 6.2.1.4.3.3 (Containment Pressure-Temperature Results) and Table 6.2.2-6a (Water Level Within the Reactor Building Following a MSLB) also state that termination of auxiliary feedwater can be accomplished in 10 minutes and that the 30-minute response time was used only to show conservatism in the Containment P/T Analysis. These statements remain in the Current FSAR for Callaway Plant.

The 10 minute isolation time was previously submitted for NRC review:

> Bechtel letter BLSE-2422 provides discussion on “Secondary System Pipe Ruptures Inside Containment” as a proposed revision to PSAR Section 6.2.1.3.10. Section 6.2.1.3.10.4 (Auxiliary Feedwater) states that manual isolation of the auxiliary feedwater system is assumed at 600 seconds (10 minutes). This assumption was modeled into original containment analyses and incorporated into PSAR, Rev. 13.
>
> The Auxiliary Feedwater System (FSAR Section 10.4.9) was submitted to the NRC for review against the Standard Review Plan, via letter SLNRC 81-39. The 10-minute operator action was included in Section 10.4.9.2.3.

NRC Question 2:

Describe the assumptions used in the CONTEMPT containment calculations. Discuss if the assumptions included the temperature flash assumption and if the Tagami and Uchida heat transfer correlations were used. Address what conservatisms are included in the calculation of structural heat sink areas and coatings, and in the value of the containment volume.

AmerenUE Response:

The calculation of heat sink areas were reduced by either construction tolerances or a 10% factor for those items that did not have known tolerances. The containment volume listed in Callaway's FSAR of 2.5E6 ft^3 is the value used in the analysis. This value is approximately 4% lower than the calculated containment free volume. The Uchida heat transfer correlation is used in the analysis of MSLB cases. Callaway's limiting MSLB cases are insensitive to changes in the flashing modeling. This is because limiting pressure-temperature results are produced by split breaks which have no entrained moisture.

NRC Question 3:

Explain how the generic calculations of mass and energy release can be independent of feedwater (FW) assumptions and discuss if this is related to the discussion of Section 3.1.5 of WCAP-8822.

AmerenUE Response:

The generic mass and energy release values provided by Westinghouse do not account for plant specific values. Plant specific values regarding feedwater assumptions were incorporated into the original Bechtel analysis as part of the steam generator dryout time calculation. This is the method discussed in Safety Analysis Standard 12.2, Section III.D. Safety Analysis Standard 12.2 is Appendix A of WCAP 8822. The revised main feedwater isolation time and auxiliary feedwater flowrate were used to calculate the new dryout time.

NRC Question 4:

It is stated that the peak containment pressure calculated by CONTEMPT remains below the Bechtel analysis for a steam line break with an increase in stroke time of the MFIVs and for an increase in the maximum auxiliary feedwater (AFW) flow due to the ARC valves when the cases are considered separately. This is stated in the last paragraph of the Main Steam Line Break Mass and Energy Release Analysis section (starting page 21 of 33) of Attachment 2. Discuss an verify that when both effects are considered together the peak containment pressure remains below the Bechtel analysis for a steam line break.

AmerenUE Response:

Both effects were considered and analyzed together. The original Bechtel Pressure-Temperature results remain bounding when the combined effects of the increased AFW flowrate and increased MFIV isolation time are considered together.

B. Regarding SGTR Accident Re-analysis (TAC No. MB9876)

B.1 Human Factors Considerations

NRC Question 1:

The paragraph just below the table states that "Simulated control room exercises were performed in 2003 for this accident. The exercises have demonstrated that the operator action times that serve as inputs to the thermal-hydraulic analysis have increased above the times originally analyzed in the SGTR with overfill analysis presented to the NRC in [letter] ULNRC-1518, dated May 27, 1987 ..." The sentences in the application may mean that the times in the previous analysis were increased to those listed in the table solely to meet the increased operator action times observed in simulated control room exercises in 2003. Discuss this and explain what the times in the table are based upon.

AmerenUE Response:

During the review and screening of the Feedwater Isolation Valve (MFIV) Actuator modification, it was determined that changing the MFIV isolation time from 5 to 15 seconds had the potential to adversely impact the SGTR-Overfill analysis. In the process of determining the sensitivity of Overfill consequences to the MFIV isolation time, it was identified that the assumed Operator action times used as inputs in the SGTR-Overfill analysis had not been maintained as currently valid.

The issues related to the validity of the SGTR-Overfill inputs were entered into the Callaway Plant's corrective action program. Additionally, a Licensee Event Report (LER 2003-003-00) was submitted to report this issue.

A complete re-analysis of the SGTR-Overfill sequence was performed. Re-validation of all analysis inputs was performed as a part of this re-analysis effort. During the re-analysis effort it was necessary to establish a new set of operator action times for use in the re-analysis. A series of simulator exercises were performed during Spring of 2003 to establish the new operator action times. Additionally, AmerenUE and Westinghouse personnel reviewed the operator action times used by other Westinghouse plants to benchmark the new Callaway times to ensure that the new times were reasonable.

The analysis effort was then completed based on the new set of operator action times. These are the values provided in our License Amendment Request.

As will be discussed below, all operating crews have demonstrated that they are capable of satisfying these times.

NRC Question 2:

The NRC staff's evaluation dated March 30, 1987, of the Westinghouse Owners group WCAP-10698, "SGTR Analysis Methodology to Determine the Margin to Steam Generator Overfill," stipulated plant-specific criteria for assessing operator action times in the event of an SGTR.

Address the criteria as updated below:

AmerenUE Response:

It should be noted that the Callaway SGTR analysis is not performed using the methodology described in WCAP-10698. The Callaway SGTR analysis is based on the SNUPPS methodology which was originally developed for Callaway and Wolf Creek.* The treatment of operator response times used in the SGTR analysis is discussed as follows:

a. Provide simulator and emergency operating procedure training related to a potential SGTR.

AmerenUE Response:

Callaway's Licensed Operator Training program provides training on the SGTR accident sequence and the associated emergency operating procedures (EOPs) used to respond. All Callaway Licensed Operators have been trained on the updated Procedure E-3 "Steam Generator Tube Rupture", using both classroom and simulator training sessions.

b. Using typical control room staff as participants in demonstration runs, show that the operator action times assumed in the SGTR analysis are realistic and achievable.

AmerenUE Response:

All Callaway Plant operating crews have demonstrated that they can achieve the new SGTR-Overfill operator action times. This includes both on-shift and staff crews.

c. Complete demonstration runs to show that the postulated SGTR accident can be mitigated within a period of time compatible with overfill prevention, using design basis assumptions regarding available equipment and its impact on operator response times. All control room crews should demonstrate a response time which is less than the operator response time assumed in the analysis for the accident.

AmerenUE Response:

Overfill prevention is not demonstrated for Callaway. The analysis and operator action times are commensurate with mitigation of the consequences of an overfill event.* All Callaway Plant operating crews have demonstrated that they can achieve the new SGTR-Overfill operator action times. This includes both on-shift and staff crews.

* The sentences in **bold** are related to NRC Question 3 in Section B.1 of Enclosure 4.

d. Describe the means the emergency operating procedures specify for identifying the steam generator (SG) with the ruptured tube, provide the expected time period for determining that SG, and discuss the effects on the duration of the accident.

AmerenUE Response:

E-0 "Reactor Trip or Safety Injection" is the initial procedure used following initiation of the accident sequence. The first time-critical diagnostic steps are those related with the transition from E-0 to E-3 "Steam Generator Tube Rupture." Diagnostic methods used by E-0 include:

- Process Radiation Monitors
- Sampling and Laboratory Analysis
- Uncontrolled Increase in Narrow Range Level for any Steam Generator

Exclusive reliance on sampling and laboratory analysis would result in delaying operator response to a large SGTR such as the Licensing Bases case. During the simulator exercises discussed previously that demonstrated that all crews could achieve the new analysis times, the Licensed Operators based their E-0 to E-3 transition on process radiation monitors and behavior of steam generator narrow range level indication.

Identification of the steam generator with the ruptured tube occurs following the E-0 to E-3 transition. Step 2 of E-3 provides the procedural diagnostic guidance to identify the ruptured steam generator. Identification of the ruptured steam generator is based on:

Unexpected increase in any SG narrow range level

OR

High activity in any SG sample

OR

High radiation from any SG Steamline
(This step would require a local Health Physics Technician to perform radiation surveys)

OR

High activity in any SG blowdown line sample.

As discussed earlier, exclusive reliance on laboratory analysis or local surveys would result in delaying operator response to a large SGTR such as the Licensing Bases case. During the simulator exercises discussed previously, that demonstrated that all crews could achieve the new analysis times, the Licensed Operators identified the ruptured steam generator based on behavior of the narrow range level.

Identification of the ruptured steam generator is not a step specifically modeled in the analysis. Therefore, the timing of this identification is not firmly established.

Identification of the ruptured steam generator would occur prior to the rapid cooldown step which is assumed to occur at 30 minutes.

All crews demonstrated their capability to achieve the new assumed time values using the diagnostic methods specified by Callaway EOPs E-0 and E-3.

NRC Question 3:

Below is an additional question based on the following statements given in the licensee's responses to the questions above and shown in **bold** above in Section B.1 of Enclosure 5:

> "It should be noted that the Callaway SGTR analysis is not performed using the methodology described in WCAP-10698. The Callaway SGTR analysis is based on the SNUPPS methodology which was originally developed for Callaway and Wolf Creek."
>
> "Overfill prevention is not demonstrated at Callaway. The analysis and operator action times are commensurate with mitigation of the consequences of an overfill event."

Discuss why the SGTR analysis and operator actions are set for mitigation of the consequences of an overfill event instead of being set for prevention of the overfill event (i.e., WCAP-10698).

AmerenUE Response:

WCAP-10698 has never been a licensing basis for Callaway Plant. The SGTR analysis forming the licensing basis for Callaway Plant is the SNUPPS methodology that was originally developed for Callaway and Wolf Creek. The WCAP was only mentioned in our response because it was cited as the basis for the NRC question. SGTR analyses for Callaway Plant demonstrate that overfill prevention is not possible, therefore, the analysis and operator action times are commensurate with mitigation of the consequences of an overfill event.

B.2

NRC Question 1:

The licensee did not address the impact of the proposed changes on the ability of the control room habitability systems to maintain doses to operators within the criteria of 10 CFR Part 50, Appendix A, General Design Criteria 19. The staff notes that the analysis of record overfill case thyroid dose (pre-incident spike) at the EAB was 24 rem and that this had increased to 46 rem in the new overfill case. Given the significant increase in the EAB dose, the staff suspects that the control room dose would have similarly increased. Respond to Question a or b below as applicable.

a. Provide a description of the assumptions, inputs, methods and results of the evaluation that demonstrates that GDC-19 will continue to be met.

b. If the licensee has not evaluated the control room dose but is relying on the dose being bounded by that determined for another accident, provide a justification that addresses the considerations in Paragraphs 7a through 7d of RIS 2001-19, as applicable.

In either case, please provide the information requested in item 1.a of Generic Letter 2003-01, as it applies to the SGTR with overfill, in your response. If you have already docketed your response to GL 2003-01, provide a citation to that response.

AmerenUE Response:

The submitted changes to the SGTR Overfill analysis do not adversely affect Callaway's Licensing Bases Control Room radiological consequences.

Callaway's Licensing Bases is that LOCA provides the limiting radiological consequence to Control Room personnel. As a result, the only Control Room radiological consequences reported in Callaway's FSAR are for the LOCA sequence. As a result, the analysis efforts were directed towards identifying the case that produces maximum offsite consequences.

Several offsite dose cases were performed by Westinghouse to identify the case that produces maximum offsite dose. The maximum offsite dose case is based on SI at initiation of the accident sequence. If SI occurs at initiation of the accident sequence, then the assumptions used in the LOCA analysis regarding the time of Control Room isolation are valid for the SGTR Overfill with SI at accident initiation. For this case, SGTR Overfill Control Room consequences are bounded by the FSAR reported value for LOCA.

The delayed SI cases analyzed by Westinghouse determined that SI would occur at approximately 6 minutes into the accident sequence. This is prior to the start of relief from the ruptured steam generator. Relief begins at approximately 11 minutes. Therefore, for the purposes of Control Room radiological consequences analyses, it is valid to assume that the Control Room would be isolated prior to the post-accident release of radioactivity to the environment.

Appendix F of SLNRC 86-01 describes the radiological methods used to calculate radioactivity releases to the atmosphere and offsite doses for SGTR events. One of the assumptions specified is that 100% of the iodine contained in the fraction of the break flow to the faulted SG that flashes upon reaching the secondary side is conservatively included even when the RETRAN analysis shows that no steam is released from the second side atmospheric steam dumps or safety valves. This assumption was intended to conservatively maximize offsite doses. It was not intended to imply that radioactivity release via the flashing pathway would occur prior to Control Room isolation.

The SGTR-Overfill release rates are less than those found in the LOCA analysis. The Control Room would be isolated prior to the initiation of release. Therefore, it is our evaluation that the SGTR-Overfill would not adversely affect the Licensing Bases post-accident Control Room consequences analysis contained in Callaway's FSAR.

With regards to items 7a through 7b of RIS 2001-19:

a. The control room design is often optimized for the DBA LOCA, and the protection afforded for other accident sequences may not be as advantageous. For example, in most designs, control room isolation is actuated by engineered safety feature (ESF) signals such as containment high pressure or safety injection (SI), or radiation monitors, or both. For accidents that rely on radiation monitor actuation, there may be a time delay in isolation that would not occur for the immediate SI signal that would result from a LOCA. In such cases, contaminated air would enter the control room for a longer period preceding isolation than it would for a LOCA.

AmerenUE Response:

This was discussed in the above paragraphs.

b. The configuration of radiation monitors has an impact on their sensitivity. Ideally, the radiation monitors would be located outside in air ventilation intake ductwork. However, there are system designs that place the radiation monitor in recirculation ductwork or downstream of filters. There are also designs that use area radiation monitors. In these latter designs, the contaminated air continues to build up in the control room volume until the concentration is large enough to actuate the radiation monitor.

AmerenUE Response:

GK-RE-04/05 located in the intake ductwork. They are not downstream of filters or located in recirculation ductwork.

c. In some cases, control room radiation monitor setpoints may have been based on external exposure concerns, for example, 2.5 mrem/hour, rather than thyroid dose from inhalation. The airborne concentration of radioiodines will likely cause elevated thyroid doses before reaching the concentration of all radionuclides necessary to alarm the monitor. This condition is typically seen with accidents that involve a high iodine-to-noble-gas ratio, such as main steam line breaks in PWRs.

AmerenUE Response:

These radiation monitors are not relied on. Control Room isolation is initiated prior to the initiation of the release of radioactivity.

d. The distance between the control room and the release point, and the associated wind sectors, may be different for each postulated accident. These differences are usually not significant with regard to offsite doses, but may be significant for control room assessments because of the shorter distances typically involved. The X/Q for the DBA LOCA may not be applicable to other DBAs. A ground-level release associated with a non-LOCA event may be more limiting than the elevated release associated with LOCAs at plants with secondary containments or enclosure buildings.

AmerenUE Response:

The Callaway FSAR does not include Control Room X/Q values for the main steam safety valves. However, we believe that use of the Reactor Building X/Q is an appropriate alternative. The Reactor Building X/Q does not credit any elevation. It is based on ground-level release assumptions.

NRC Question 2:

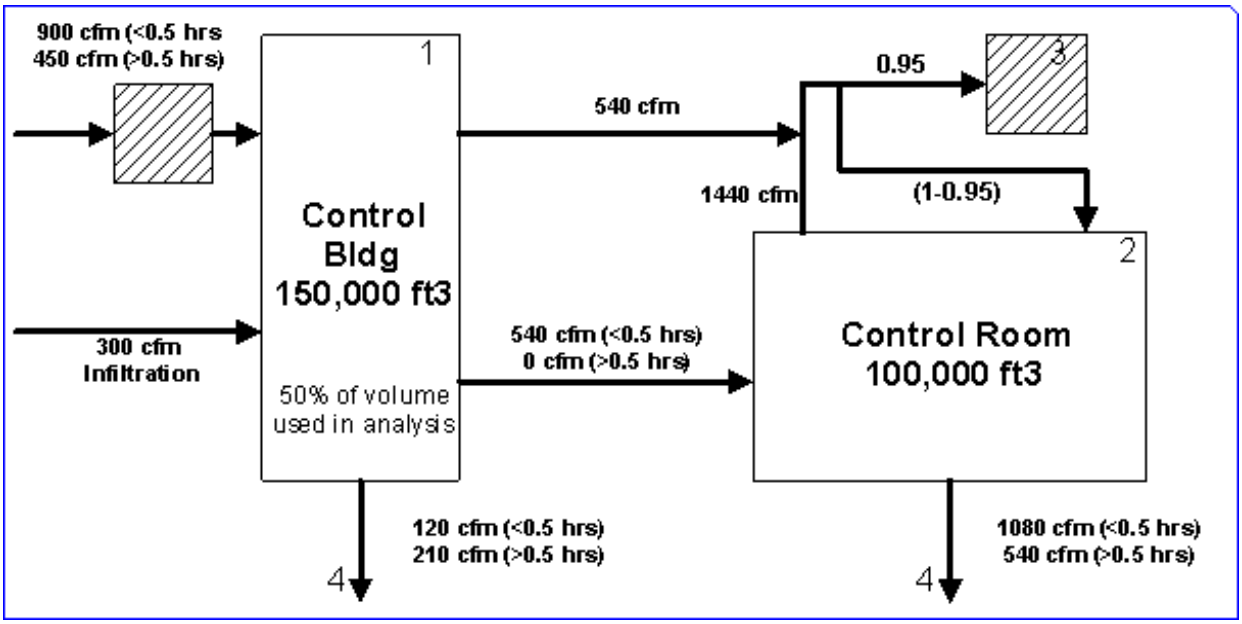
Figures 15.6-3P, 15.6-3.2d, and 15.6-3.2h of the submittal provide data for the intact steam generators (SGs). Discuss if the data represents each intact SG or the total for all intact SGs.

AmerenUE Response:

The intact SG figures represent a lumped SG that is representative of the 3 intact SGs.

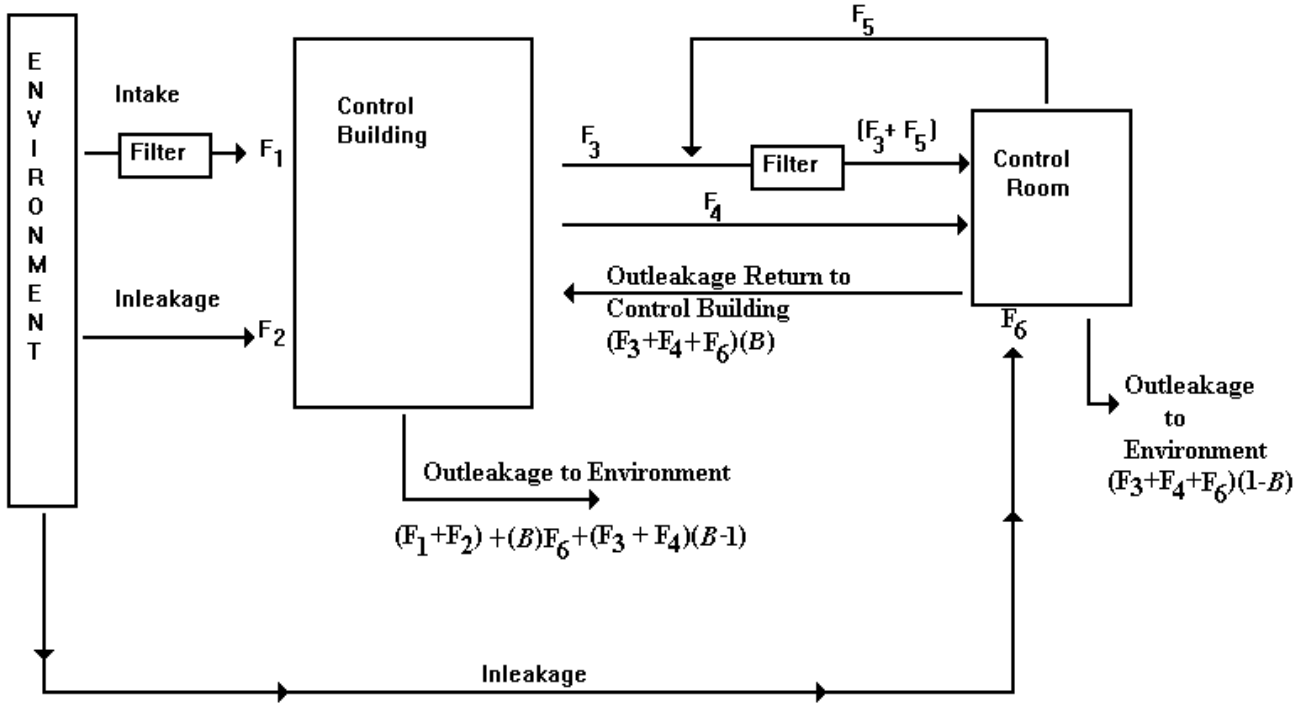
NRC Question 3:

The figure below represents the staff's interpretation of appropriate modeling of your control building and control room. Node 3 is the recirculation filter; node 4 is a "sink." Discuss the staff's interpretation and confirm if the staff's understanding is correct. In particular, discuss if the expected re-alignment will occur at 30 minutes for this event and provide the basis for this conclusion.



<u>AmerenUE Response</u>:

The staff's flow diagram does not match the one used by AmerenUE. The following diagram is taken from Section 15A of the Callaway FSAR:



The diagram has two flow paths not shown on the staff's diagram. The first is direct inleakage from the environment to the Control Room. The second represents Control Room outleakage returning to the Control Building.

Additionally, the flow values for the F_3, F_4, and F_5 flow paths shown in the staff's diagram are incorrect. These values are as follows:

F_3 440 cfm
F_4 440 cfm
F_5 1360 cfm

These values are based on the following:

The Control Room makeup flow from the Control Building is established at 400 cfm per train. The tolerance on this is +/- 10%. Higher flow values for the F_3 and F_4 flow paths produces more limiting results. Therefore, a value of 440 cfm is used in the analysis for these flow paths.

Total flow through the Control Room recirculation filter is limited to 2000 cfm. The tolerance on this is +/- 10%. Using the minimum value of 1800 cfm produces more limiting results. This 1800 cfm represents the sum of F_3 and F_5. Therefore, a value of 1360 is used for F_5. This minimizes the recirculation cleanup rate and provides conservative results.

NRC Question 4:

Discuss (1) the iodine appearance rates for I-131 to I-135, in Ci/hr, to which the multiplier of 335 will be applied and (2) the assumed duration of the accident induced spike.

AmerenUE Response:

The iodine appearance rates are based on maximum allowable RCS DEI-131 levels and maximum letdown cleanup rates. The accident initiated spike is assumed to last for 8 hours.

NRC Comment:

In reviewing the licensee's application, the staff had the following comments on statements made by the licensee. First, with regard to the conclusion that 10 CFR 50.67 applied to this amendment, the staff believes that the requirements of 10 CFR 50.67 do not apply to this amendment request. This is based on the definition of "source term" in 10 CFR 50.2 and the statements of consideration for the final 10 CFR 50.67 rule (63 FR 71990 dated December 23, 1999). Second, the licensee has requested staff approval to use Regulatory Guide (RG) 1.195 for other licensing basis dose applications. The staff believes that 10 CFR 50.59 already provides the licensee with an adequate mechanism to implement the guidance of RG 1.195. The guide provides methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations; however, the guide also contains alternative methods that must be considered on a case-by-case. The staff considers blanket approval of the use of RG 1.195 would confer approval for each of these case-by-case situations, most of which are not considered relevant to the technical specification changes requested or to the re-analysis of the SGTR with overfill. If the licensee believes that the staff has misunderstood its rational in these two areas, the licensee should provide further explanation of its position.

AmerenUE Response:

AmerenUE concurs that 10 CFR 50.59 provides a mechanism to implement the guidance of Regulatory Guide 1.195. The discussion in the submittal regarding 10 CFR 50.59 and the application of Regulatory Guide 1.195 is included to support the expectation that the review and approval for the use of the 335 spiking factor would be based on AmerenUE's specific application, even though NRC's approval for other utilities gives precedence for our request for NRC review and approval.

REVISED AMERENUE RESPONSE TO NRC COMMENT AFTER QUESTION 4

NRC Question 4:

Discuss (1) the iodine appearance rates for I-131 to I-135, in Ci/hr, to which the multiplier of 335 will be applied and (2) the assumed duration of the accident induced spike.

AmerenUE Response:

The iodine appearance rates are based on maximum allowable RCS DEI-131 levels and maximum letdown cleanup rates. The accident initiated spike is assumed to last for 8 hours.

NRC Comment:

In reviewing the licensee's application, the staff had the following comments on statements made by the licensee. First, with regard to the conclusion that 10 CFR 50.67 applied to this amendment, the staff believes that the requirements of 10 CFR 50.67 do not apply to this amendment request. This is based on the definition of "source term" in 10 CFR 50.2 and the statements of consideration for the final 10 CFR 50.67 rule (63 FR 71990 dated December 23, 1999). Second, the licensee has requested staff approval to use Regulatory Guide (RG) 1.195 for other licensing basis dose applications. The staff believes that 10 CFR 50.59 already provides the licensee with an adequate mechanism to implement the guidance of RG 1.195. The guide provides methods acceptable to the NRC staff for implementing specific parts of the NRC's regulations; however, the guide also contains alternative methods that must be considered on a case-by-case. The staff considers blanket approval of the use of RG 1.195 would confer approval for each of these case-by-case situations, most of which are not considered relevant to the technical specification changes requested or to the re-analysis of the SGTR with overfill. If the licensee believes that the staff has misunderstood its rational in these two areas, the licensee should provide further explanation of its position.

AmerenUE Response:

The AmerenUE discussion regarding the applicability of 10CFR50.59 to the Reg. Guide 1.195 methodologies was intended to reflect the pathway used in AmerenUE's evaluation regarding whether or not the 335 iodine spiking factor could be implemented without prior NRC approval. The 335 iodine spiking factor has been approved for use by another Licensee in their SGTR analysis. Part of our bases for concluding that implementation of the 335 spiking factor required prior NRC approval involved the position that source terms used in FSAR analyses are not exclusively regulated by 10CFR50.59.

AmerenUE is requesting approval for use of the 335 spiking factor in our SGTR analyses, and to use the ICRP-30 dose conversion factors, on a forward-fitting bases for all of our FSAR Chapter 15 radiological consequences analyses.

B.3 Reactor Systems Review

NRC Question 1:

It is stated in the submittal that the SGTR re-analysis is performed with revised operator action times, inputs, and assumptions that are consistent with the current plant configuration and operation. Tabulate these changes and provide a justification for each change.

AmerenUE Response:

The following discussion provides a listing of inputs used in the new analysis. These inputs were validated to reflect current plant configuration.

The following new/revised assumptions are implemented in this new analysis:

- Initial conditions
  Vessel average temperature = 583.4°F - 5°F = 578.4°F. This reflects the Callaway Tavg band of 583.4°F - 588.4°F that is possible with a 5°F reduction in design Tavg.
  RCS flow = minimum measured flow = 382640 gpm. (prev. 374360 gpm)
  Steam generator pressure = 908 psia. (prev. 939 psia)
  Feedwater temperature = 390°F. (prev. 446°F)

- Decay heat model is changed to 80% of the 1979 ANS 2σ model used by transient analysis.

- The following AFW flow rates are modeled prior to partial/full isolation of AFW flow to the ruptured SG, to approximate the flows measured in the field:

  AFW flow to ruptured SG before isolation of turbine driven AFW pump flow to the ruptured steam generator, at the intact SG pressure of 1235.7 psia is used as a base. As the intact SG pressure drops the flow to the ruptured SG is reduced. This reduction is larger for higher ruptured SG pressures. A conservative model for this reduction is used based on the lowest ruptured SG pressure of 414.7 psia. This model is reflected in the table below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)		Intact SG Pressure (psia)	Reduction in AFW to Ruptured SG (gpm)
414.7	1317.0		414.7	72.6
614.7	1214.0		614.7	55.4
814.7	1104.0		814.7	37.8
1014.7	982.0		1014.7	20.0
1139.7	895.0		1139.7	8.6
1235.7	823.0		1235.7	0.0

AFW flow to intact SGs (total for the 3) before isolation of turbine driven AFW pump flow to the ruptured steam generator, at ruptured SG pressure of 1139.7 psia is used. As the ruptured SG pressure drops the flow the intact SGs is reduced. This reduction is small and is neglected:

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1691.0
414.7	1576.0
614.7	1455.0
814.7	1326.0
1014.7	1186.0
1139.7	1091.0
1235.7	1013.0

- The following AFW flow rates are modeled to reflect the flow network analysis results after partial/full isolation of AFW flow to the ruptured SG:

AFW flow to ruptured SG after isolation of turbine driven AFW pump flow to the ruptured steam generator is provided in the table below:

Ruptured SG Pressure (psia)	AFW to Ruptured SG (gpm)
414.7	770.
614.7	712.
814.7	651.
1014.7	586.
1139.7	537.
1235.7	498.

AFW flow to intact SGs (total for the 3) after isolation of turbine driven AFW pump flow to the ruptured steam generator, and after complete isolation of AFW to the ruptured SG is provided in the table below:

Intact SG Pressure (psia)	AFW to Intact SGs (gpm)
214.7	1760.
414.7	1656.
614.7	1546.
814.7	1425.
1014.7	1295.
1139.7	1205.
1235.7	1129.

- AFW temperature = 32°F. (prev. 70°F)
- AFW flow is initiated 5 seconds after reactor trip (to bound the field tested 12 second delay), with a 30-second ramp up to full flow.
- SI/charging flow temperature = 37°F. However, due to stability concerns the charging flow is modeled at 41°F. (prev. 50°F)
- SI action is initiated coincident with reactor trip. This was determined to provide limiting results relative to the scenario with SI actuation on low pressurizer pressure.
- MFIV isolation is initiated by the SI signal. This was determined to provide limiting results relative to the scenario with MFW isolation delayed until a high SG level signal was generated.
- MFIV closure is modeled as a step function with a 17 second delay.
- Operator actions modeled [the following:]

  Isolation of turbine driven AFW flow to the ruptured SG at 10 minutes from the start of the event.

  Isolation of all AFW flow to the ruptured SG at 20 minutes from the start of the event.

  Initiate cooldown by dumping steam from the lumped intact loop SG ARV after 30 minutes from reactor trip (which is at the start of the event).

  The cooldown is terminated when the core outlet temperature reaches the target temperature specified in the EOPs as a function of the ruptured SG pressure. The current EOP cooldown target temperature table (assuming normal containment and RCS temperatures per incore thermocouples) is:

SG Pressure (psig)	Target Temperature (˚F)
> 1200	535
1100 - 1199	524
1000 - 999	512
900 – 999	499
800 – 899	485
700 – 799	470
600 – 699	450
500 – 599	430
430 - 499	410

  Initiate depressurization by dumping steam using pressurizer relief valve at a time such that the depressurization will be completed at 40 minutes from the start of the event.

  SI flow is terminated 5 minutes after the depressurization is completed. This is approximately 45 minutes from the start of the event.

  Depressurize by dumping steam using pressurizer relief valve following SI termination to terminate break flow at 60 minutes from the start of the event.

  Cooldown to RHR cut in is initiated after break flow is terminated.
- The break flow model is adjusted to model all flow at the enthalpy of the ruptured steam generator outlet header.
- The break flow flashing fraction is calculated assuming all break flow is at the ruptured loop hot leg temperature.
- The initial conditions are based on 10% steam generator tube plugging. However, the SG heat transfer model in RETRAN is based on 15% tube plugging and this is conservatively

retained. A conservatively high initial secondary mass is assumed to bound 0% tube plugging.

- MSIV isolation on reactor trip and the assumed loss of offsite power has not been changed, although it could be significantly delayed based on the expected operator response. Early isolation of the MSIV allows the ruptured SG to depressurize due to the addition of the (maximum) AFW flow, while the intact SG pressure stays relatively high. This results in increased break flow to the ruptured SG, which is conservative. It also leads to higher AFW flow to the ruptured SG. If the MSIV would be left open, the SGs would tend to be at the same pressure, which would be closer to that of the intact SGs (which are lumped together in the RETRAN model). Also, with the MSIV open overfilling the ruptured SG would not necessarily lead to water relief, since the water could go to the intact SG. The secondary pressure would not spike and the safety valve would not lift.

The time to close the MSIV is left at 1.5 seconds. As noted above early isolation is considered to be more limiting.

NRC Question 2:

Discuss the computer codes used in the thermal-hydraulic analysis of the SGTR event. If any of the computer codes are not approved by the NRC, provide a justification for their use in the analysis.

AmerenUE Response:

The thermal-hydraulic analysis was performed using RETRAN, which is an NRC approved code.

NRC Question 3:

Provide any revised emergency operating procedure (EOP) steps (in E-2 and E-3) that are related to the actions required for isolation of auxiliary feedwater (AFW) flow to the failed steam generator (SG).

AmerenUE Response:

Emergency Operating Procedure E-3, Steam Generator Tube Rupture, was revised in support of the SGTR with overfill re-analysis. Procedure E-3 was revised so that the required operator action times provide a more aggressive response to the SGTR with overfill event. The revision included:

- Incorporation of Westinghouse recommended revisions which deleted a redundant step, reordered steps, and allow some parallel actions in order to improve the timely completion of operator actions
- Addition of several attachments to enhance the operating crew's ability to perform concurrent actions while progressing through the procedure
- Added and modified procedure notes to aid the crew in timely step completion

- Modified a step to isolate AFW flow to the ruptured SG in the case of the failed open AFW flow valve.

E-3 procedure changes were made with full validation on the simulator and were included in the training for all licensed operators.

NRC Question 4:

For the case of an AFW control valve failing to its open position, discuss what operator actions are needed to isolate the failed SG from continued AFW flow injection (i.e., what backup valve needs to close and its location), including where these actions are performed.

AmerenUE Response:

Once the operator has identified the need to isolate AFW flow to the ruptured steam generator, he will attempt to close the flow control valves from the main control board. If this is unsuccessful, the operator has two options to complete the isolation. If adequate AFW flow can be maintained without the AFW pump associated with the flow to the ruptured steam generator, that pump may be stopped from the main control board. The alternative is to dispatch a local equipment operator to manually isolate AFW flow to the ruptured steam generator.

NRC Question 5:

Discuss the integrity of the main steam line for SG overfill (i.e., with the entire main steam line up to the MSIV filled with water).

AmerenUE Response:

A pipe stress analysis of the most highly stressed main steam line, which assumed the main steam line up to the MSIV was filled with water, was performed by Bechtel. This analysis found the piping and supports still met ASME Code Class 2 requirements and were acceptable for this condition.

NRC Question 6:

For the main steam line break (MSLB), discuss the effect of the proposed longer closure time (i.e., 15 seconds) of the main feedwater isolation valves (MFIVs).

AmerenUE Response:

As discussed in the submittal, the increase in MFIV stroke time potentially affects the main steam line break (MSLB) mass and energy (M&E) releases inside and outside containment.

The analysis of the MSLB M&E releases outside containment (FSAR Section 3B.4.2) assumes main feedwater isolation coincident with reactor trip, with no delays associated with instrumentation or valve stroke. This is a conservative assumption for this event. Quicker isolation of main feedwater flow produces more limiting Main Steam Tunnel pressure-temperature results due to minimized total mass addition to the SGs and resultant higher levels of superheat in the blowdown Mass and Energy (M&E) releases. Therefore, an increase in the MFIV closure time does not adversely impact this analysis.

The proposed increase in MFIV isolation time affects the key parameter of steam generator dry-out time in the MSLB inside containment analysis. This parameter is addressed in the original pressure-temperature calculations. Although slower valve closure time impacts mass and energy releases in general, the proposed stroke time increase of 10 seconds does not specifically impact the original calculated Mass and Energy (M&E) Releases. The original portion of the analysis is unaffected by the proposed change in MFIV isolation time.

The analysis of the MSLB M&E releases inside containment (FSAR Section 6.2.1.4) for Callaway is limiting at part-power conditions, resulting from a split rupture in a steam line. The analysis supporting the limiting MSLBs for containment response is a generic calculation performed by the NSSS supplier for the Model F steam generator design. The MSLB M&E releases for split ruptures are generic with no specific assumptions regarding time for main feedwater isolation (as well as other critical protection functions). It has been confirmed that the generic assumptions made in the original Mass and Energy Releases analysis bound the Callaway Plant proposed MFIV stroke time of 15 seconds.

Adjustments have been made to the generic M&E release values for specific Callaway Plant conditions. As previously stated, the key parameter affected by longer MFIV isolation time is steam generator dry-out time.

Post-accident steam generator dry-out is defined as the time when flow into the affected generator is equal to flow out of the generator, after the break has occurred. (In order to reach dry-out, the initial inventory must be depleted; break flow is then a function only of flow into the generator. Following dry-out, the magnitude of the break flow is not influenced by the secondary side water inventory). If dry-out occurs after the termination of AFW flow to the faulted steam generator, the mass release rate is set to zero following dry-out. If dry-out occurs prior to AFW termination, the mass release rate is set to the AFW flow rate. The mass release rate is then subsequently set to zero once AFW flow is terminated.

The proposed increase in MFIV stroke time would result in additional main feedwater mass being introduced into the affected steam generator. The additional mass would then be released to containment, which would delay dry-out of the affected generator. This would then provide the potential to lead to higher post-MSLB pressures or temperatures inside containment.

AmerenUE has performed a calculation to quantify the impact of the additional 10 seconds of main feedwater flow to the steam generators following initiation of the accident sequence. This calculation quantified the additional steam generator secondary side mass inventory, following a MSLB inside containment. Then CONTEMPT computer code runs were executed to determine the impact of the additional mass on post-MSLB containment pressures and temperatures. These CONTEMPT runs found that the proposed MFIV actuator replacement

and associated increase in MFIV stroke time caused no adverse impact on post-MSLB containment pressures and temperatures. The calculated pressure and temperature profiles remain bounded by the analysis envelopes originally calculated by Bechtel.

NRC Question 7:

Following the SG overfill, it is assumed that the safety valve associated with the failed SG is stuck open with an effective flow area equal to 5% of the total safety valve flow area. Discuss the basis for the assumed effective flow area value.

AmerenUE Response:

This flow area value was mandated as part of Question 1 from NRC staff requests for additional information transmitted to Union Electric in an NRC letter dated 11/12/86 from P. W. O'Connor to D. F. Schnell. Union Electric committed to use this value which was documented in our response to the NRC and transmitted in ULNRC 1518, dated May 27, 1987.

NRC Question 8:

In the sequence of events for a SGTR with overfill, it is indicated that the operator actions to terminate AFW flow from the turbine driven AFW (TDAFW) pump to the failed SG will be completed within 10 minutes following the event initiation. Discuss why these actions are expected to be completed within 10 minutes while the operator actions to isolate AFW flow from MDAFW pumps are expected to be completed within 20 minutes.

AmerenUE Response:

The SGTR with overfill is a licensing bases event which includes a single failure. The single failure is the MDAFP flow control valve. The failed MDAFP flow control valve produces more limiting results than a failed TDAFP valve. So within 10 minutes, the operator diagnoses the need to isolate AFW to the ruptured generator. The TDAFP valve is assumed to respond and function properly when the RO closes it from the Main Control Board. The MDAFP flow control valve is the assumed failure, so the additional 10 minutes allows time to either dispatch a local operator, or to turn off the MDAFP.

NRC Question 9:

Figure 15.6-3P, "Feedwater Flow Rate," of the proposed FSAR page changes, does not appear to be consistent with the time assumed for closure of the MFIVs (i.e., 15 seconds). Discuss how the FSAR figure is consistent with the assumed MFIV closure time.

AmerenUE Response:

Figure 15.6-3P, "Feedwater Flow Rate", is only included in Attachment 6 of the submittal in order to show the placement of new Figures associated with the SGTR with overfill accident

which is being added into the FSAR. Figure 15.6-3P is not associated with the SGTR with overfill accident, but the SGTR with stuck-open ASD. The SGTR with stuck-open ASD is the current FSAR accident Analysis of Record as the most limiting in terms of radiological consequences. The SGTR with stuck-open ASD analysis is not impacted by the increase in MFIV isolation time and was not re-analyzed. The current analysis for SGTR with stuck-open ASD is more limiting in the assumption of MFIV isolation time. As discussed in the Callaway FSAR, Section 15.6.3.2.g, the analysis assumes the feedwater isolation signal occurs 2.3 seconds after reactor trip and the feedwater isolation valves stroke closed within 2.0 seconds. These are considered the minimum expected delay and stroke time, respectively, which decreases heat removal from the reactor coolant system resulting in higher reactor coolant system temperatures and pressures. This results in maximum flashed fraction and break flow in the analysis. In summary, quicker isolation of main feedwater flow causes an increase in break flow flashing and steam releases, resulting in more radioactivity released to the atmosphere and higher radiological consequences. As a result, the proposed increase in MFIV stroke time does not invalidate the results of the current FSAR SGTR with stuck-open ASD analysis.

NRC Question 10:

Discuss why the change to the MFIV closure time from 5 seconds to 15 seconds does not affect other events for which re-analyses of these event should be performed to support the proposed Technical Specification changes.

AmerenUE Response:

As discussed in the license submittal, a complete review of FSAR Chapter 6.2 and Chapter 15 accident analyses was performed to evaluate the impact of the increase in MFIV closure time from 5 seconds to 15 seconds. In some of these accident analyses, the MFIV isolation time is not explicitly modeled or MFIV position is not an essential analysis consideration. These accident sequences or categories of sequences are listed in the submittal and are eliminated from further consideration for impact based on the increase in MFIV isolation time.

The remaining accident analyses were reviewed and evaluated for impact. These accidents are also listed in the submittal. Individual summaries are provided for each accident to discuss the evaluation and the results. In each case, the overall impact is determined to have little or low significance and re-analysis of the accident is not warranted. The accident scenario for SGTR with overfill is the exception. A complete re-analysis was performed for the SGTR with overfill accident.

AmerenUE is confident that the SGTR with overfill is the only accident scenario significantly impacted by the increase in MFIV closure time from 5 seconds to 15 seconds. The SGTR with overfill was re-analyzed in support of the proposed change.

C. Regarding Opening in Secondary Shield Wall (TAC No. MB9879)

C.1

The following is an RAI for the license amendment request application dated June 27, 2003, related to an opening in the secondary shield wall.

NRC Questions:

On review of the Callaway drawings (in the FSAR and the additional drawing provided by the licensee to the NRC project manager), it has been identified that the proposed opening in the shield wall, which is depicted in drawing no. C-2S2977, may not be consistent with the following statement by the licensee on page 5, second paragraph, of Attachment 1 to the application: "In order to preclude radiation streaming and dose resulting from creating the opening in the secondary shield cubicle wall, alternative shielding will be applied to the opening and access control entryway to limit radiation doses consistent with maintaining them as low as [is] reasonably achievable (ALARA)." The alternative shielding indicated on the drawings does not appear to "preclude streaming" for the following reasons:

1. The shield does not cover the entire height of the opening. Approximately four square feet of opening is unshielded.

2. No shielding is provided on the top, or the deck, of the security cage to prevent radiation (penetrating the secondary shield wall opening at a up or down slant angle) from streaming into accessible spaces above and below the security cage.

3. The "alternative" lead shield that is included in the design will provide only approximately three orders of magnitude less shielding (one tenth thickness of lead versus about four tenth thicknesses of concrete).

Therefore, it is requested that the following be provided to clarify the application:

1. Explain how the alternative shielding is as effective as the original shield design, or

2. a) Calculate the expected increase in radiation dose rates outside the secondary shield cubicle resulting from this modification.

2.b) Estimate the increase in dose expected for workers accessing these areas during periods of reactor shutdown and power operations and show that this is ALARA.

2.c) Verify that the modification does not create a very high radiation area, as defined in 10 Part 20, or describe the "additional measures" required by 10 CFR 20.1602.

AmerenUE Response:

First, our choice of words "in order to preclude radiation streaming" is incorrect. When we were in the development stage of this modification we were looking at using some kind of shield door rather than a labyrinth wall we are now proposing. The door would have "precluded streaming";

however, it would have weighed over 4000 pounds and was considered a personnel safety hazard to open and close it every time someone entered or exited the area. It was also our further concern that, because the door was so heavy, people would just leave it open during plant shutdown periods. With the door open we would have no shielding in the area just outside of the wall. During periods of plant shutdown, there could be several people working in the area which would have a higher dose than what is there before the modification. By using the labyrinth wall, sufficient shielding would be provided during plant shutdown periods to prevent this for occurring and also eliminate the personnel safety concern with handling a very heavy door.

The primary reason for cutting the access door is to make access to the sludge lance platform for all four steam generators and all four reactor coolant pumps faster, safer, and more efficient resulting in a lower total dose exposure to personnel. Currently, in order to get to the sludge lance platforms, one must enter the bio-shield area from the 2000 foot elevation, climb a ladder to the primary steam generator platform, then climb another ladder to the secondary platform. During Steam Generator Maintenance activities, access via the primary steam generator platform is not available due to high dose rates and contamination levels in the area forcing personnel to climb a temporary scaffold ladder to the platform. Extra dose is received installing and removing this access ladder as well as the additional time just to get to this location. The door through the bio-shield wall coupled with the elevated walkway connecting the two sludge lance platforms will eliminate the need for this temporary ladder and access time.

During normal plant operations, we understand that dose rates in the zone of influence will be higher than they are before the modification. The dose rates in the affected zone on the 2026 elevation will increase to about 67 mrem/hour during power operations with the labyrinth shield in place and will be insignificant during periods of plant shutdown. We also agree that the labyrinth shield wall was insufficient to cover the opening and would allow exposure to personnel in some areas of the affected zone. We will be increasing the height of the wall by 2’-0" to assure the doorway is covered. In addition, we will be adding approximately 12 inches to the bottom of the shield to provide shielding to feet and lower legs of personnel passing through the area. The area affected on the 2026 elevation of containment is small and there is no sensitive equipment in the area. The number of personnel passing through the area during power operations will be minor and for very short durations.

The dose rates in the affected zone on the 2000 elevation will increase to about 188 mrem/hour during power operations. It is understood that this is a larger increase in dose rates; however; the area affected on the 2000 foot elevation of containment is very small and there is no sensitive equipment in the area. During power operations if personnel entered the affected area, they would do so just to pass through to gain access to equipment located beyond the zone that remains shielded. In addition, the number of personnel passing through the affected area during power operations will be minor. Dose rates during plant shutdown will be insignificant because the source will be different.

The operating deck (2047’-6" elevation) is shielded by a 2’-0" thick concrete slab, coupled with the distance from the proposed opening, the increase in dose rate during power operations within the zone of influence is only approximately 3 mrem/hour and is considered insignificant.

Conclusion:

Based on the above, total exposure to plant personnel will be reduced as a result of providing an access doorway through the secondary bio-shield wall at this location. The increase in dose rates in the affected areas will not result in the creation of a very high radiation area as defined in 10 Part 20 of 10 CFR 20.1602.

D. Regarding LBB Methodology for Opening in Secondary Shield Wall (TAC No. MB9879)

D.1 Materials Branch Review:

NRC Question

Discuss the rationale for not considering the potential effect of thermal stratification, cycling and striping (TASCS) in the horizontal sections of RHR and accumulator piping discussed in the subject WCAP topical reports (provided in the application dated June 27, 2003) due to potential leakage of hot fluids past the isolation valve or heat transfer across the isolation valve, including, in particular, the startup of the systems. The discussion should also address the contribution to thermal fatigue due to TASCS in the above cases and will the nodal location of maximum stress change as a result of the above.

AmerenUE Response:

The following is the rationale for not considering the potential effect of thermal stratification, cycling and striping (TASCS) due to potential leakage of hot fluids past the valve or heat transfer across the valve in the horizontal sections of the RHR and accumulator piping. In particular, the startup of the systems is addressed.

Accumulator Piping:

The unisolable section of piping extending from the cold leg to the check valve is relatively short. This piping, including the horizontal section, will be at nearly the cold leg temperature due to turbulent penetration heating from the cold leg. Leakage from the cold leg is unlikely due to a second upstream check. Leakage would result in a rise in accumulator tank level and pressure, which would be detected. Heat transfer across the check valve is generally not significant with respect to structural considerations. Startup of the accumulator system would indicate a safety injection (accumulator blowdown). The higher flow rates preclude the possibility for stratification, as the line has full flow. The contribution to thermal fatigue due to TASCS is therefore negligible. The nodal location of maximum stress will not change.

RHR Piping:

The unisolable section of piping extending from the hot leg to the isolation valve is relatively short. This piping, including the horizontal section, will be at nearly the hot leg temperature due to turbulent penetration heating from the hot leg. Leakage, as at Genkai Unit 1 (see NRC Bulletin 88-08, Supplement 3), is possible if the isolation valve has a leakoff line. Callaway Plant has performed a modification to cap the subject leakoff line, therefore leakage is highly unlikely due to a second downstream isolation valve (which also has a capped leakoff line). Heat transfer across the isolation valve is generally not significant with respect to structural considerations. Startup of the RHR system generally occurs with the reactor coolant system much cooler and under significantly less pressure loading (approximately 350 F and about 440 psig) than for normal operating conditions (typically 618 F and 2235 psig). The higher RHR flow

rates preclude the possibility for stratification, as the line has full flow. The contribution to

thermal fatigue due to TASCS is therefore negligible. The nodal location of maximum stress will not change.

D.2 Mechanical Engineering Branch Review:

The following is a request for information on the three topical reports submitted for the leak before break (LBB) methodology in the application submitted June 27, 2003 (ULNRC-04868).

PART A - Surge Line

In accordance with Standard Review Plan (SRP) 3.6.3, "Determination of Rupture Locations and Dynamic Effects Associated with the Postulated Rupture of Piping," the following are questions related to the staff's review of Westinghouse topical report WCAP-15983-P, "Technical Justification for Eliminating Pressurizer Surge Line Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," dated February 2003.

NRC Question 1:

Provide assurance that the internal loads used in the calculations were determined based on the as-built configuration of the surge line.

AmerenUE Response:

The internal loads used in the LBB analysis are based on the as-built configuration of the surge line.

NRC Question 2:

Provide assurance that the wall thickness of all components of the surge line meets the minimum ASME Code, Section III, Class 1, wall thickness requirements.

AmerenUE Response:

All components of the surge line met the minimum ASME Code, Section III, Class 1, wall thickness requirements.

NRC Question 3:

Discuss compliance with the snubber surveillance requirements of the Callaway Technical Requirements Manual (TRM) to provide assurance that snubber failure rate are acceptably low.

AmerenUE Response:

Callaway's snubber surveillance program is controlled per FSAR Section 16.7.2.1.1 (we do not have a TRM) and representative snubber samples are tested per the criteria defined therein to ensure that failure rates are acceptably low. Sixteen mechanical snubbers are currently in service within the referenced WCAP piping system boundaries for the Pressurizer Surge Line,

12" RHR lines and 10" Accumulator lines. Six of these have been functionally tested since initial plant operation and all have passed. Six others have been hand stroked during this period and all have passed. Two more of these sixteen snubbers are scheduled for functional testing during the next refueling outage. Our snubber surveillance program will continue to provide assurance that snubber failure rates are acceptably low, as required per NRC regulations.

NRC Question 4:

Section 4.2 of the WCAP defines "TH" as the "Applicable Thermal Expansion Load (Normal or Stratified)." Provide clarification indicating if the "normal thermal expansion load" includes both axial and bending loads, and if the thermal stratification also includes axial and bending loads. Provide justification why normal and stratified thermal loads are not additive.

AmerenUE Response:

Normal thermal expansion loads include both axial and bending loads, and the thermal stratification loads also include axial and bending loads.

The normal thermal loads and stratified thermal loads are not additive because they are both thermal loads and they do not occur at the same time.

NRC Question 5:

In Section 4.2 of the WCAP, the applied loads for crack stability analysis include loads due to seismic anchor motion (SAM) and the leak before break (LBB) margin is stated to be reduced to 1.0. SRP 3.6.3 indicates that the margin of 1.4 can be reduced to 1.0 if these loads are combined absolutely with the safe shutdown earthquake (SSE) load and the individual operating loads. Provide verification that the SAM loads were combined absolutely with the SSE and the operating loads, as specified in the SRP 3.6.3.

AmerenUE Response:

There is no seismic anchor motion loads due to SSE in the surge line because the surge line was coupled with the loop in the piping stress analysis.

NRC Question 6:

In Table 4-2 of the WCAP, which shows Cases A through G for normal and faulted loading cases for LBB evaluations, provide the justification for not including normal thermal expansion loads in the load combination Case E.

AmerenUE Response:

Case D is the traditional LBB analysis faulted case with pressure, deadweight, thermal normal expansion and SSE. Case E is an additional faulted case for the surge line with pressure, deadweight, stratified thermal expansion loads during normal operation and SSE. The two thermal cases (thermal normal and stratified thermal) do not occur at the same time. Case D includes the thermal normal loads and Case E includes the stratified thermal expansion loads during normal operation.

NRC Question 7:

For Node 3030 in Table 4-4 of the WCAP, which provides a summary of LBB loads and stresses, provide a table for Cases A through G of Table 4-2 showing the individual force and moment components due to pressure, deadweight, thermal expansion, thermal stratification, SSE inertia and SSE SAM.

AmerenUE Response:

[proprietary information]

NRC Question 8:

In Section 4.5 of the WCAP, provide the minimum wall thickness at the weld counterbore used in the analysis.

AmerenUE Response:

The minimum wall thickness at the weld counterbore used in the analysis is 1.251 inches.

NRC Question 9:

Provide the basis or a reference for Equation 5-3 in Section 5.2.2 of the WCAP.

AmerenUE Response:

The Equation 5-3 was developed by Westinghouse by using the Reference 5-2 (WCAP-15983-P Revision 0) as a basis.

NRC Question 10:

In support of the statement in Section 5.2.3 of the WCAP that "The leak rates were calculated using the normal operating loads at the governing location identified in Section 4.0," provide a

representative detailed leak rate calculation, including the calculation of the crack opening area, for the postulated through-wall circumferential crack at Node 3030.

AmerenUE Response:

Information for a representative leak rate calculation:

Node 3030 Case A:

Input Data:

Pipe outer diameter= 14.00 in, thickness=1.251 in, Pressure= 2250 psia, Temperature= 653°F, Axial force= 242351 lbs, Moment=1615950 in-lbs, E=25035000 psi, Yield stress= 27230 psi.

Leak rate calculation is an iterative process involving the use of Westinghouse proprietary computer codes. First the crack opening area was calculated for a circumferential through-wall crack length utilizing the NUREG/CR-3464 (Reference 5-3 of WCAP-15983-P Revision 0) method and the input data listed above. Using the resulting crack opening area, input data listed above, assumed leakage crack length and the method described in Section 5.2.2 of WCAP-15983-P Revision 0, leak rate was calculated. This process has to be repeated for various crack lengths to determine the crack length which will yield a leak rate of 10 gpm.

For NODE 3030 Case A, using the methodology described above, the results are:
for a circumferential through-wall crack length of 3.89'', the value of crack opening area is 0.02711 in^2 and the calculated leak rate is 10 gpm.

NRC Question 11:

The fatigue crack growth analysis for the pressurizer surge line was performed at the same location where the maximum ASME Code, Section III, Class 1, cumulative usage factor was previously calculated under the effects of thermal stratification. Provide the specific location on the pressurizer surge line, and the value of the maximum cumulative usage factor used in the analysis.

AmerenUE Response:

The fatigue crack growth (FCG) analysis location is shown in Figure 6-1 of WCAP-15983-P, Revision 0. The value of the maximum ASME Code, Section III, Class 1, cumulative usage factor previously calculated under the effects of thermal stratification is 0.4.

NRC Question 12:

In Section 6.2 of the WCAP, the term "stress cuts" is used in the statement that "Fatigue crack growth analyses were carried out along five stress cuts...." Discuss what a "stress cut" is and provide an explanation of the manner in which it is used in the fatigue crack growth analyses.

AmerenUE Response:

Figure 6-2 of WCAP-15983-P shows the five locations (stress cuts) where the fatigue crack growth analyses were performed. A stress cut is an angular location on the cross-section of the pipe.

At each location stresses for each transient were obtained along with the number of cycles for input into the fatigue crack growth analyses.

PART B - Accumulator Lines

For WCAP-16019-P "Technical Justification for Eliminating 10" Accumulator Lines Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," the questions are the following.

NRC Question 1:

Provide the ASME Code, Section III, Class 1, cumulative usage factors at Nodes 3020 (Accumulator Loop 2), 3120 (Accumulator Loop 2), and 3295 (Accumulator Loop 3).

AmerenUE Response:

The ASME Code, Section III, Class 1, cumulative fatigue usage near Node 3020 (Accumulator Loop 2) is [proprietary] and near Node 3120 (Accumulator Loop 2) is [proprietary]. At Node 3295 (Accumulator Loop 3) no ASME Code, Section III, Class 1, cumulative fatigue usage was calculated since it is located in the Class 2 portion of the line.

NRC Question 2:

Provide assurance that the internal loads used in the calculations were determined based on the as-built configuration of the accumulator lines.

AmerenUE Response:

The loads used in the LBB analysis are based on the as-built configuration of the accumulator lines.

NRC Question 3:

Provide assurance that the wall thicknesses of all components of the accumulator lines meet the minimum ASME Code, Section III, Class 1, wall thickness requirements.

AmerenUE Response:

All components of the accumulator lines met the minimum ASME Code, Section III, Class 1, wall thickness requirements.

NRC Question 4:

Discuss compliance with the snubber surveillance requirements of the Callaway Technical Requirements Manual (TRM) to provide assurance that the snubber failure rate on the accumulator lines is acceptably low.

AmerenUE Response:

Callaway's snubber surveillance program is controlled per FSAR Section 16.7.2.1.1 (we do not have a TRM) and representative snubber samples are tested per the criteria defined therein to ensure that failure rates are acceptably low. Sixteen mechanical snubbers are currently in service within the referenced WCAP piping system boundaries for the Pressurizer Surge Line, 12" RHR lines and 10" Accumulator lines. Six of these have been functionally tested since initial plant operation and all have passed. Six others have been hand stroked during this period and all have passed. Two more of these sixteen snubbers are scheduled for functional testing during the next refueling outage. Our snubber surveillance program will continue to provide assurance that snubber failure rates are acceptably low, as required per NRC regulations.

NRC Question 5:

In Section 4.2 of the WCAP, the applied loads for crack stability analysis include loads due to the safe shutdown earthquake (SSE). Provide verification that SSE loads include both inertia and seismic anchor motion loads, and that these are combined absolutely, as specified in the Standard Review Plan (SRP) 3.6.3.

AmerenUE Response:

There are no seismic anchor motion loads due to SSE in the accumulator lines because the accumulator lines were coupled with the loop in the piping stress analysis.

NRC Question 6:

Provide detailed justification for not including operating basis earthquake (OBE) loads in the fatigue crack growth analysis

AmerenUE Response:

In the LBB analysis through-wall flaws are postulated and it was demonstrated that LBB condition exists with ample margins for the Callaway accumulator lines. For the Fatigue Crack Growth (FCG) assessment surface flaws were postulated and demonstrated that the postulated

surface flaws will not grow though the wall considering the transients and cycles for the design life of the plant. Table 6-2 of WCAP-16019-P Revision 0 shows the results of the Fatigue Crack Growth (FCG) assessment. Even for the largest postulated flaw of 0.3 inch, this is about 33% of the wall thickness; results show that the growth is only about 10%. It was concluded that the flaw growth through the wall will not occur during the 40 year design life of the plant. OBE is an upset condition and it was judged that OBE will have insignificant impact on the FCG results and conclusions. Therefore OBE was not considered in the FCG assessment. Note: For the ASME Code, Section III, Class 1, cumulative usage fact ors calculation the contribution from OBE was small.

PART C - RHR Lines

For WCAP-16020-P "Technical Justification for Eliminating 12" Residual Heat Removal (RHR) Rupture as the Structural Design Basis for the Callaway Nuclear Power Plant," the questions are the following.

NRC Question 1:

Provide the ASME Code, Section III, Class 1, cumulative usage factors at Nodes 3285 (RHR Line Loop 1) and 3020 (RHR Line Loop 4).

AmerenUE Response:

The ASME Code, Section III, Class 1, cumulative fatigue usage near Node 3285 (RHR Line Loop 1) is [proprietary] and near Node 3020 (RHR Line Loop 4) is [proprietary].

NRC Question 2:

Provide assurance that the internal loads used in the calculations were determined based on the as-built configuration of the residual heat removal (RHR) lines.

AmerenUE Response:

The loads used in the LBB analysis are based on the as-built configuration of the RHR lines.

NRC Question 3:

Provide assurance that the wall thicknesses of all components of the RHR lines meet the minimum ASME Code, Section III, Class 1, wall thickness requirements.

AmerenUE Response:

All components of the RHR lines met the minimum ASME Code, Section III, Class 1, wall thickness requirements.

NRC Question 4:

Discuss compliance with the snubber surveillance requirements of the Callaway TRM to provide assurance that snubber failure rate are acceptably low.

AmerenUE Response:

Callaway's snubber surveillance program is controlled per FSAR Section 16.7.2.1.1 (we do not have a TRM) and representative snubber samples are tested per the criteria defined therein to ensure that failure rates are acceptably low. Sixteen mechanical snubbers are currently in service within the referenced WCAP piping system boundaries for the Pressurizer Surge Line, 12" RHR lines and 10" Accumulator lines. Six of these have been functionally tested since initial plant operation and all have passed. Six others have been hand stroked during this period and all have passed. Two more of these sixteen snubbers are scheduled for functional testing during the next refueling outage. Our snubber surveillance program will continue to provide assurance that snubber failure rates are acceptably low, as required per NRC regulations.

NRC Question 5:

In Paragraph 4.2, the applied loads for crack stability analysis include loads due to SSE. Provide verification that SSE loads include both inertia and seismic anchor motion loads, and that these are combined absolutely, as specified in the SRP 3.6.3.

AmerenUE Response:

There is no seismic anchor motion loads due to SSE in the RHR lines because the RHR lines were coupled with the loop in the piping stress analysis.

NRC Question 6:

Provide detailed justification for not including OBE loads in the fatigue crack growth analysis

AmerenUE Response:

In the LBB analysis through-wall flaws are postulated and it was demonstrated that LBB condition exists with ample margins for the Callaway RHR lines. For the Fatigue Crack Growth (FCG) assessment surface flaws were postulated and demonstrated that the postulated surface flaws will not grow though the wall considering the transients and cycles for the design life of the plant. Table 6-2 of WCAP-16020-P Revision 0 shows the results of the Fatigue Crack Growth (FCG) assessment. Even for the largest postulated flaw of 0.35 inch, this is about 35% of the wall thickness; results show that the growth is only about 4%. It was concluded that the flaw growth through the wall will not occur during the 40 year design life of the plant. OBE is an upset condition and it was judged that OBE will have insignificant impact on the FCG results and conclusions. Therefore OBE was not considered in the FCG assessment. Note: For the ASME Code, Section III, Class 1, cumulative usage factors calculation the contribution from OBE was small.

LIST OF ACRONYMS

AFW	Auxiliary feedwater
ALARA	As low as is reasonably achievable
ARC	Automatic recirculation control check
ASME	American Society of Mechanical Engineers
FSAR	Callaway Final Safety Analysis Report
EOP	Emergency operating procedures
ESF	Engineered safety feature
FMEA	Failure modes and effects analysis
FW	Feedwater
FWLB	Feedwater line break
FCG	Fatigue crack growth
LBB	Leak before break
LOAC	Loss of alternating current (AC) power
LONF	Loss of normal flow
MFIV	Main feedwater isolation valve
MFW	Main feedwater
MSLB	Main steam line break
OBE	Operating basis earthquake
PWR	Pressurized water reactor
RCS	Reactor coolant system
Refuel XX	Refueling Outage No. XX
RHR	Residual heat removal
SAM	Seismic anchor motion
SG (or S/G)	Steam generator
SGTR	Steam generator tube rupture
SRP	NRC Standard Review Plan
SSE	Safe shutdown earthquake
TASCS	Thermal stratification, cycling, and striping
TRM	Technical Requirements Manual
WCAP	Westinghouse topical report

ADDITIONAL QUESTIONS RAISED BY THE NRC STAFF

Technical Review Areas:

A. MFW/AFW Modifications (TAC No. MB9875)

A.1, Plant Systems Review

No additional questions.

A.2, Containment Safety Assessment Review:

1. Topical Report WCAP 8822 states that the described methods are applicable to Model D steam generators, and, "with minor alterations," to Model 51 steam generators. However, Callaway uses Model F steam generators which the FSAR states are "similar in configuration" to the Model 51 steam generators.

Describe the differences between the Callaway steam generators and the Model 51 steam generators and why the calculations of the topical report apply given those differences.

2. Please verify that there were no changes to the containment calculation methods and input described in the FSAR and that used for the proposed license amendment in the June 27, 2003 letter. For example, Table 6.2.1-2 through Table 6.2.1-5.

B. Regarding SGTR Accident Re-analysis (TAC No. MB9876)

B.1, Human Factors Considerations:

No additional questions.

B.2, Accident Radiological Consequences:

In the response to the staff RAIs, you provided an alternative figure from the FSAR for the control room/control building/environment flow interactions. Although the response identifies two flow paths not shown on the staffs figure, insufficient data is provided for the staff to determine the flow rate in these two paths: Please arrange a telecon regarding the following additional questions.

(1) what is the value of the direct inleakage to the control room (F6)? Is this equal to F2?

(2) What is the value of the constant "B" shown in the figure for determining outleakage from the control room returning to the control building? This constant is identified as the

fraction of outleakage that returns to the control building. What is this constant based upon? Does the same value apply to the other two outleakage expressions as well?

B.3, Plant Systems Review:

No additional questions.

C. Regarding Opening in Secondary Shield Wall (TAC No. MB9879)

No additional questions.

D. Regarding LBB Methodology for Opening in Secondary Shield Wall (TAC No. MB9879)

D.1, Materials Branch review:

No additional questions.

D.2, Mechanical Engineering Branch Review:

No additional questions.

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